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ARKANSAS NUCLEAR ONE
UNIT 1

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APPENDIX A
TECHNICAL SPECIFICATIONS

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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.

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1 DEFINITIONS

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

1.1 RATED POWER

Rated power is a steady state reactor core output of 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 140F). 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 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 - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

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.

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 shall be defined by the following equation and is expressed as a percentage.

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

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 FIRE SUPPRESSION WATER SYSTEM

The fire suppression water system consists of: water sources, pumps, and distribution piping with associated sectionalizing isolation valves. Such valves include the hose standpipe shutoff valves and the first valve ahead of the water flow alarm device or each sprinkler system.

1.9 STAGGERED TEST BASIS

A staggered test basis shall consist of:

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

b. The testing of one system, subsystem, train or designated component at the beginning of each subinterval.

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1.10 Dose Equivalent I-131

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

1.11 Liquid Radwaste Treatment System

A Liquid Radwaste Treatment System is a system designed and used for holdup, filtration, and/or demineralization of radioactive liquid effluents prior to their release to the evironment.

1.12 Purge - Purging

Purge or Purging is the controlled process of discharging air or gas from a confinement to reduce the airborne radioactivity concentration in such a manner that replacement air or gas is required to purify the confinement.

1.13 Member(s) of the Public

Member(s) of the Public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

1.14 Exclusion Area

The exclusion area is that area surrounding ANO within a minimum radius of .65 miles of the reactor buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

1.15 Unrestricted Area

An unrestricted area shall be any area beyond the exclusion area boundary.

1.16 Core Operating Limits Report

The CORE OPERATING LIMITS REPORT is the ANO-1 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification 6.12.3. Plant operation within these operating limits is addressed in individual specifications.

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 when the reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The maximum local fuel pin centerline temperature shall be $\leq 5080 - (6.5 \times 10^{-3} \times$ (Burnup, MWD/MTU)°F) for TACO2 applications and $\leq 4642 - (5.8 \times 10^{-3} \times$ (Burnup, MWD/MTU)°F for TACO3 applications. Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

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 could 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 a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

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A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure for the allowable RC pump combination has been considered in determining the Variable Low RCS Pressure-Temperature Protective Limits.

The Variable Low RCS Pressure-Temperature Protective Limits presented in the COLR represent the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps which is based on the nuclear power peaking factors (3) as specified in the COLR with potential fuel densification effects.

The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by the limiting combination of the radial peak, axial peak, and position of the axial peak.

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.

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 flow rates for the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop.

The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive of all possible reactor coolant pump maximum thermal power combinations as specified in the COLR. The Variable Low RCS Pressure-Temperature Protective Limits in the COLR represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line, the Variable Low RCS Pressure-Temperature Protective Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for less than four reactor coolant pumps operating of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR 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 BAW-2 or the BWC correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

For each combination of operating reactor coolant pumps of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump combination. The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

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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. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3%. However, if found outside of a ±1% tolerance band, they shall be reset to 2500 psig ±1%. 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 2450 psig. ([4])

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.11.1

(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 the Protection System Maximum Allowable Setpoints for Axial Power Imbalance as given in the COLR.

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 setpoints plus calibration and instrumentation errors.

Nuclear Overpower

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

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9 percent of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is 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 1.30 (BAW-2) or 1.18 (BWC) should a low flow condition exist due to any electrical malfunction.

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The power level trip setpoint 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 setpoint 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.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. 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 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 the Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR are produced. The power-to-flow ratio reduces the power level trip associated reactor power-to-reactor power imbalance boundaries by the value specified in the COLR 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 1.30 (BAW-2) or 1.18 (BWC) by tripping the reactor due to the loss of reactor coolant

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pumps(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Table 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 (COLR) trip setpoint shown in Table 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.([2,3])

To account for the calibration and instrumentation errors, the accident analysis used the protective limit specified in the COLR.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Table 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 setpoint 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. A nuclear overpower trip set point of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

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

The purpose of the 1720 psig high-pressure trip setpoint is to prevent normal operation with part of the reactor protection system bypassed. This high-pressure trip setpoint is lower than the normal low-pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The overpower trip setpoint of ≤5.0 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

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[d] (Nominal Operating Power, 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear Power based on flow[b] and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig. min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

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

(b) Reactor coolant system flow, %

(c) The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

(d) Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.

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3. LIMITING CONDITIONS FOR OPERATION

3.0 LIMITING CONDITION FOR OPERATION (GENERAL)

3.0.1 The Limiting Conditions for Operation requirements shall be applicable during the REACTOR OPERATING CONDITIONS or other conditions specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, no further actions need be taken.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated Action requirements, within one hour action shall be initiated to place the unit in an OPERATING CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the Action requirements, the Action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

3.0.4 Entry into a REACTOR OPERATING CONDITION or other specified condition shall not be made when the conditions of the Limiting Conditions for Operation are not met and the associated action requires a shutdown if they are not met within a specified time interval. Entry into a REACTOR OPERATING CONDITION or other specified condition may be made in accordance with Action requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to REACTOR OPERATING CONDITIONS as required to comply with Action requirements. Exceptions to these requirements are stated in the individual specification.

LIMITING CONDITION FOR OPERATION (continued)

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE and (2) all of its redundant systems(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE; or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in an OPERATING CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in :

1. At least HOT STANDBY within the next 6 hours.

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in Cold Shutdown or Refueling Shutdown.

BASES

3.0.1 through 3.0.4 Establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shutdown the reactor or follow any remedial Action permitted by the Technical Specification until the condition can be met."

3.0.1 Establishes the Applicability statement within each individual Specification as the requirement for when (i.e., in which operational modes or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The Action requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of Action requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the Action requirements. In this case, conformance to the Action requirements provides an acceptable level of safety for unlimited continued operation as long as the Action requirements continue to be met. The second type of Action requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to

restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these Actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a mode or condition in which the Specification no longer applies. It is not intended that the shutdown Action requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the Action requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the Action requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual Specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the Action requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with Action requirements, the plant may have entered a mode in which a new specification becomes applicable. In this case, the time limits of the Action requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

3.0.2 Establishes that noncompliance with a Specification exists when the requirements of the Limiting Condition for Operation are not met and the associated Action requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the Action requirements within the specified time interval constitutes compliance with a Specification and (2) completion of the remedial measures of the Action requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated Action requirements.

3.0.3 Establishes the shutdown Action requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated Action requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown mode when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its Action requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower modes of operation permit the shutdown to proceed in a controlled and orderly

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manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant systems and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the Action requirements are completed, the shutdown may be terminated. The time limits of the Action requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the Action requirements have been met or the time limits of the Action requirements have not expired, thus providing an allowance for the completion of the required Actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN condition when a shutdown is required during the POWER mode of operation. If the plant is in a lower mode of operation when a shutdown is required, the time limit for reaching the next lower mode of operation applies. However, if a lower mode of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable mode, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower mode of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the Action requirements, if compliance with the Action requirements for one specification results in entry into a mode or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of Action requirements for a higher mode of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower mode of operation.

The shutdown requirements of Specification 3.0.3 do not apply in COLD SHUTDOWN and REFUELING SHUTDOWN, because the Action requirements of individual specifications define the remedial measures to be taken.

3.0.4 Establishes limitations on mode changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher mode of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the Action requirements if a change in modes were permitted. The purpose of this specification is to ensure that facility operation is not

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initiated or that higher modes of operation are not entered when corrective action is being taken to obtain compliance with a Specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with Action requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a mode change. Therefore, in this case, if the requirements for continued operation have been met in accordance with the requirements of the specification, then entry into that mode of operation is permissible. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with Action requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower mode of operation. For the purpose of compliance with this specification the term 'shutdown' is defined as a required reduction in the REACTOR OPERATING CONDITION.

3.0.5 Delineates what additional conditions must be satisfied to permit operation to continue when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of the Limiting Condition for Operation for the normal or emergency power source, not the individual Limiting Condition for Operation statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.7.2.C provides for a 7 day out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to

be consistent with the Limiting Condition for Operation statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 3.7.1.A requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. Specification 3.7.2.B provides a 24 hour out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits would also be inoperable. This would dictate invoking the applicable Limiting Condition for Operation statements for each of the applicable LCOs. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the Limiting Condition for Operation statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, shutdown is required in accordance with this specification.

During Cold Shutdown and Refueling Shutdown, Specification 3.0.5 is not applicable and thus the individual Action statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

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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. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours with the reactor critical.

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. With no reactor coolant pumps or decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system.

3.1.1.2 Steam Generator

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F.

3.1.1.3 Pressurizer Safety Valves

A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

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. The provisions of Specification 3.0.3 are not applicable.

3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.

3.1.1.5 Reactor Coolant Loops

A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:

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1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours.

B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

3.1.1.6 Decay Heat Removal

With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition, at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:*

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

B. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition.

3.1.1.7 Reactor Coolant System Vents.

At least one reactor coolant system vent path consisting of at least two valves in series shall be operable at each of the following locations whenever the Reactor Coolant average temperature is above 280°F.

1. Reactor Vessel head
2. Pressurizer steam space
3. Reactor coolant system Hot Leg high points (2 locations)

A. With one of the above vent paths inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed; restore the inoperable vent path to operable status within 30 days, or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

B. With two or more of the above vent paths inoperable, maintain the inoperable vent paths closed and restore at least two vent paths to operable status within 72 hours or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

BASES:

The plant is designed to operate with both reactor coolant loops and at least one reactor coolant pump per loop in operation, and maintain DNBR above 1.30 (for the BAW-2 correlation) and 1.18 (for the BWC correlation) during all normal operations and anticipated transients.(1)

Whenever the reactor coolant average temperature is above 280°F, single failure considerations require that two loops be operable.

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 setpoint shall be 2,500 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. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3 percent. However, if found outside the ±1 percent tolerance band, they shall be reset to 2500 psig ±1 percent.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internal vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

The reactor coolant vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head, the reactor coolant system highpoints, and the pressurizer steam space ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are covered in Section 4.0 for the class 2 valves and Table 4.1-2 for the vent paths. These are consistent with ASME Section XI and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," 11/80.

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

Enclosure 2

Bases Change of 9-15-95

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, under the provisions of 3.1.2.3, and to ASME Code limits when no fuel assemblies are present provided the reactor coolant system limits are to the right of and below the limit line in Figure 3.1.2-1. The provisions of Specifications 3.0.3 are not applicable.

3.1.2.2 Leak Tests

Leak tests required by Specification 4.3 shall be conducted under the provision of 3.1.2.3. The provisions of Specification 3.0.3 are not applicable.

3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-2 and Figure 3.1.2-3, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-2. The heatup rates shall not exceed those shown in Figure 3.1.2-2.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the right of and below the limit line in Figure 3.1.2-3. Cooldown rates shall not exceed those shown in Figure 3.1.2-3.

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 100F.

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

3.1.2.6 With the limits of Specifications 3.1.2.3 or 3.1.2.4 or 3.1.2.5 exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg to less than 200F, while maintaining RCS temperature and pressure below the curve, within the following 30 hours.

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3.1.2.7 Prior to reaching thirty one effective full power years of operation, Figures 3.1.2-1, 3.1.2-2 and 3.1.2-3 shall be updated for the next service period in accordance with 10CFR50, Appendix G. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543(5). The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.8. The provisions of Specification 3.0.3 are not applicable.

3.1.2.8 The updated proposed technical specifications referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period.

3.1.2.9 With the exception of ASME Section XI testing and when the core flood tank is depressurized, during a plant cooldown the core flood tank discharge valves shall be closed and the circuit breakers for the motor operators opened before depressurizing the reactor coolant system below 600 psig.

3.1.2.10 With the exception of ASME Section XI testing, fill and vent of the reactor coolant system, emergency RCS makeup and to allow maintenance of the valves, when the reactor coolant temperature is less than 262°F, the High Pressure Injection motor operated valves shall be closed with their opening control circuits for the motor operators disabled.

3.1.2.11 The plant shall not be operated in a water solid condition when the RCS pressure boundary is intact except as allowed by Emergency Operating Procedures and during System Hydrotest.

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 rates satisfy 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 major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 (4). The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543. (5) The chemical composition of the limiting weld material is reported in the B&W Report, BAW-2121P (6). The effect of neutron irradiation on the RT_{NDT} of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 (7).

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the thirty first effective full power year of operation. The service period was reduced by one effective full power year from that assumed in FTI Document 77-1258569-01 to be conservative with respect to independent calculations performed by the NRC staff. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all allowed operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

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.

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

Specification 3.1.2.9 is to ensure that the core flood tanks are not the source for pressurizing the reactor coolant system when in cold shutdown.

Specification 3.1.2.10 is to ensure that high pressure injection is not the source of pressurizing the reactor coolant system when in cold shutdown. The LTOP enable temperature has been calculated in accordance with Code Case N-514. Instrument error is not included in the reactor coolant temperature of 262°F.

Specification 3.1.2.11 is to ensure that the reactor coolant system is not operated in a manner which would allow overpressurization due to a temperature transient.

REFERENCES

(1) FSAR, Section 4.1.2.4

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

(3) FSAR, Section 4.3.11.5

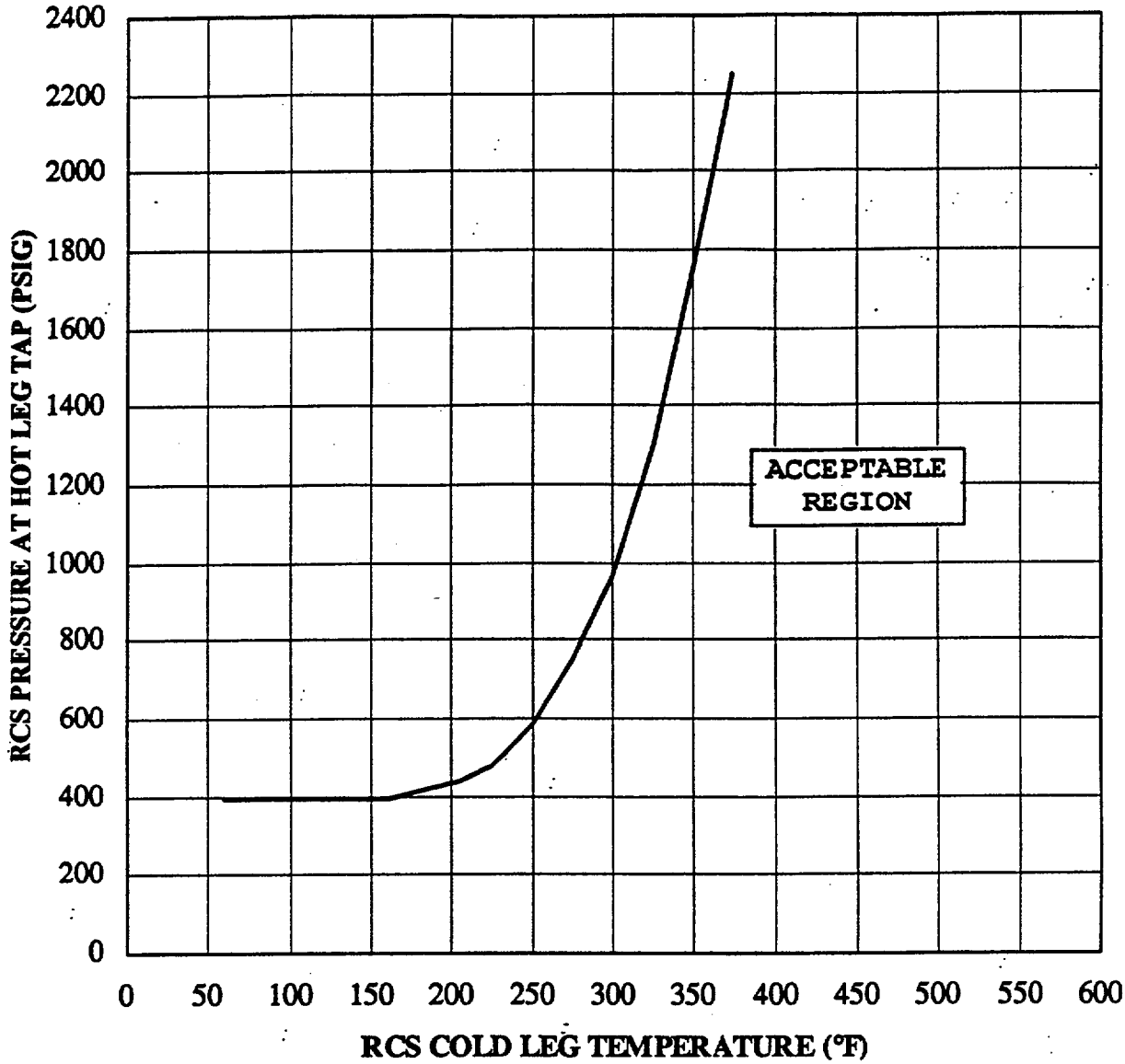
(4) FTI Document Number 77-1258569-01

(5) BAW-1543, latest revision

(6) BAW-2121P

(7) FTI Calculation Numbers 32-1245917-00 and 32-1257716-00

FIGURE 3.1.2-1
RCS INSERVICE HYDROSTATIC TEST H/U & C/D LIMITS TO 31 EFPY



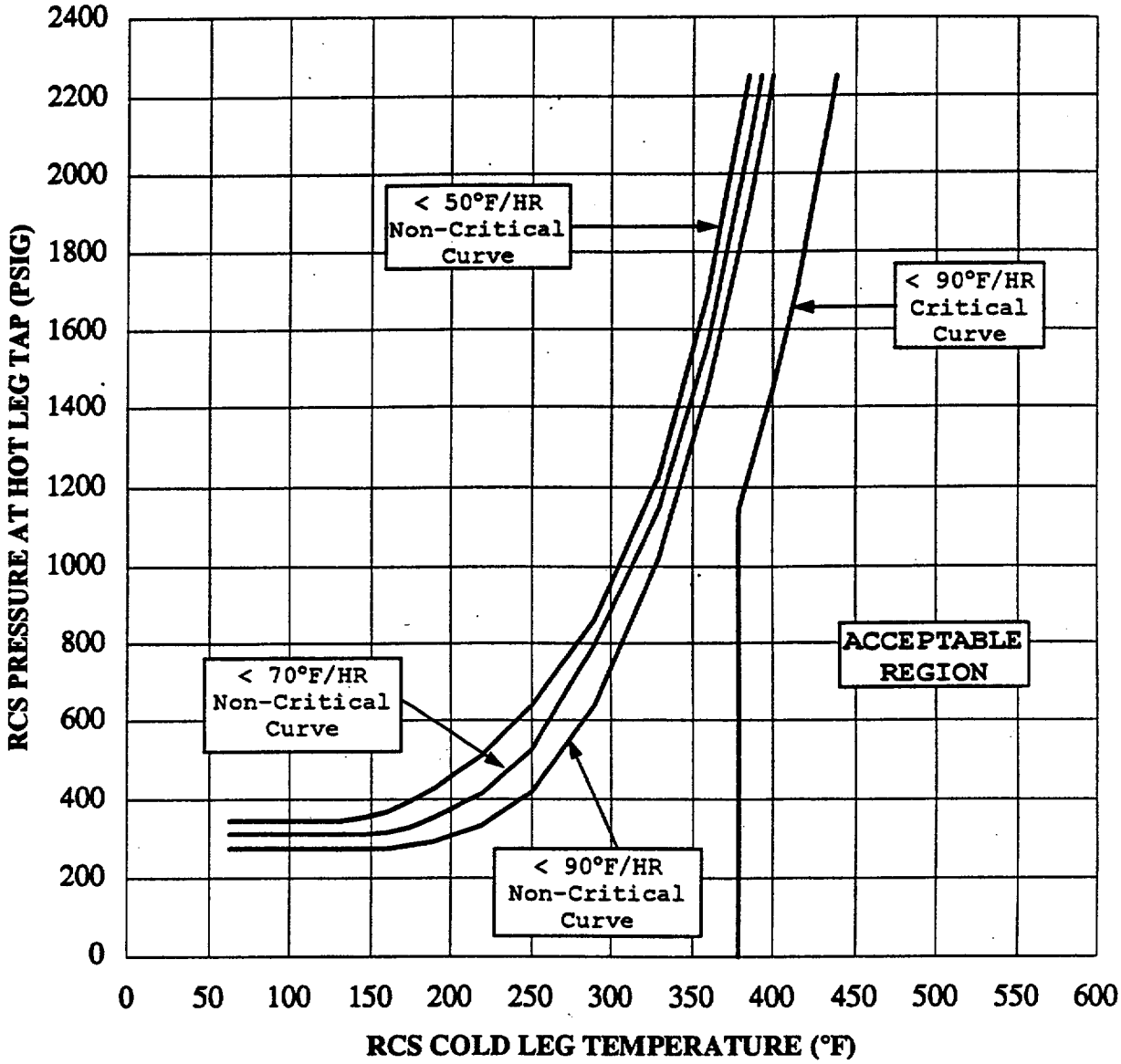
Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.

2. All Notes on Figure 3.1.2-2 are applicable for heatups. This curve is based on a heatup rate of < 90°F/HR.

3. All Notes on Figure 3.1.2-3 are applicable for cooldowns.

FIGURE 3.1.2-2
RCS HEATUP LIMITATIONS TO 31 EFPY



Notes:

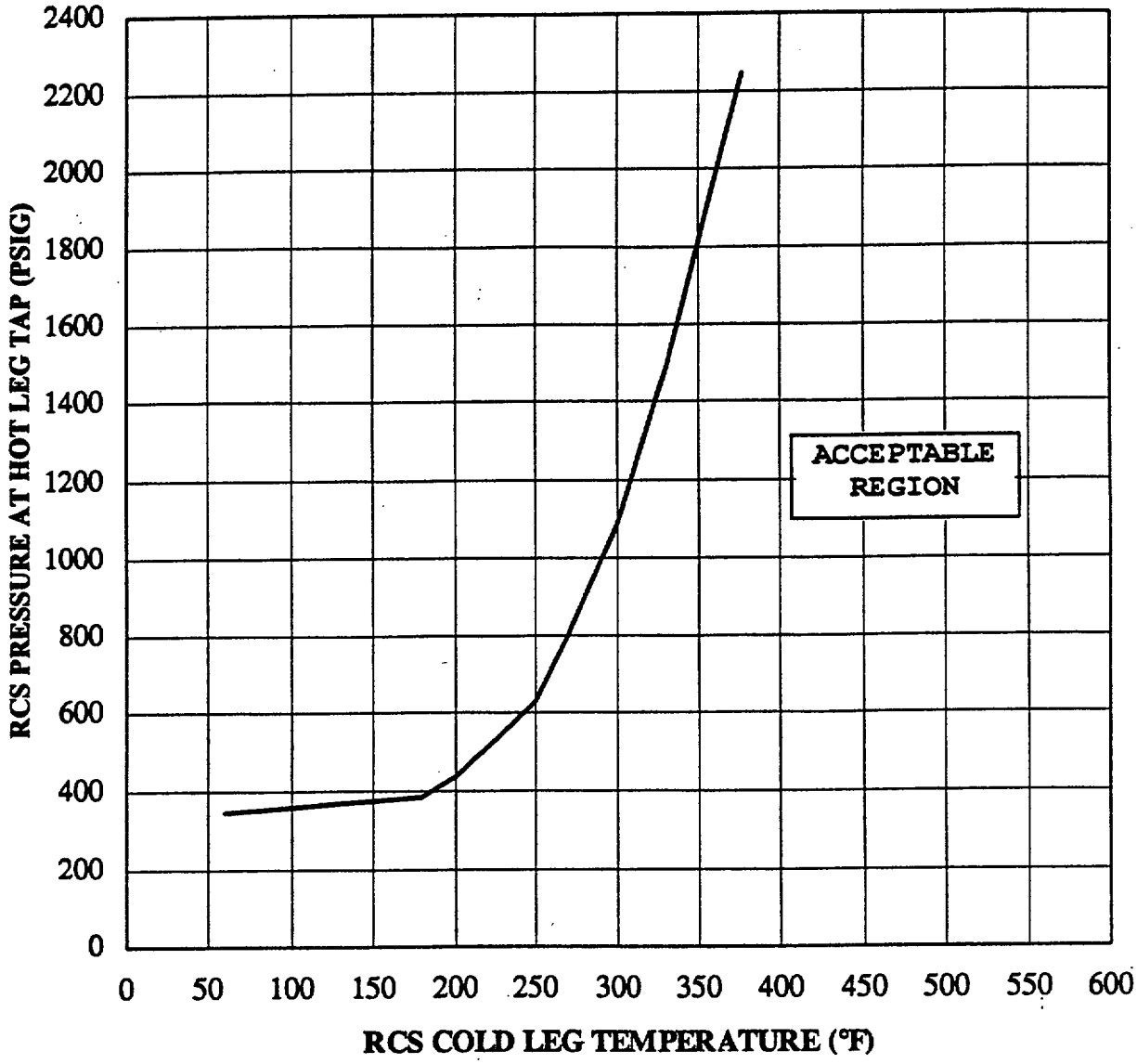
1. These curves are not adjusted for instrument error and shall not be used for operation.
2. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 300°F	None
300°F ≥ T ≥ 225°F	≤ 3
225°F > T ≥ 84°F	≤ 2
T < 84°F	No RCPs operating

4. Allowable Heatup Rates:

RCS TEMP	H/U RATE
60°F < T ≤ 84°F	≤ 15°F/HR
T > 84°F	As allowed by applicable curve

FIGURE 3.1.2-3
RCS COOLDOWN LIMITS TO 31 EFPY



Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25°F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 255°F	None
150°F ≤ T ≤ 255°F	≤ 2 (See Note 5)
T < 150°F	No RCPs operating

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280°F	100°F/HR	≤ 50°F in any 1/2 HR
280°F > T ≥ 150°F	50°F/HR (See Note 5)	≤ 25°F in any 1/2 HR
T < 150°F	25°F/HR	≤ 25°F in any 1 HR

5. If RCPs are operated < 200°F, then the RCS cooldown rate from 150°F ≤ T ≤ 180°F is reduced to 30°F in 15 hours.

3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525F 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 to the right of the criticality limit of Figure 3.1.2-2.

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, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality.

3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours.

3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in 15 minutes or be in at least Hot Shutdown within the next 15 minutes.

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 525F 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 525F 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 the limits of Figure 3.1.2-2 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 one (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 2 of the 3 emergency-powered pressurizer heaters be operable provides assurance that sufficient heater capacity ($\geq$ 126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.

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.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

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 Whenever the reactor is operating under steady-state conditions, the following conditions shall be met.

a. The total specific activity of the primary coolant shall not exceed $72/\bar{E}$ μCi/gm where $\bar{E}$ is the sum of the average beta energy and average gamma energy per disintegration in MEV/disintegration.

b. The I-131 dose equivalent of the radioiodine activity in the primary coolant shall not exceed 3.5 μCi/gm.

c. If the radioactivity in the primary coolant exceeds the limits given above, corrective action shall be taken immediately to return the coolant activity to within these specifications. If the specific activity limits given above cannot be achieved within 24 hours, the reactor shall be brought to a hot shutdown condition using normal operating procedures. If the coolant radioactivity is not reduced to acceptable limits within an additional 48 hours, the reactor shall be brought to a cold shutdown condition and the cause of the out-of-specification operation ascertained.

Bases

Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system. 2

The parameters assumed in the dose analysis for the single steam generator tube failure included the following values:

1) total primary coolant volume (mass) = 5.2×10^5 lbs.

2) total secondary coolant volume (mass) = 2×10^6 lbs.

3) leakage rate from primary to secondary system = 1 gpm.

4) fission product decay heat energy for 1 hour = 1.56×10^8 BTU.

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5) steam mass released to environs = 2.84×10^5 lbs.

6) primary coolant released to secondary (34 minutes) = 8.7×10^4 lbs.

7) minimum primary to secondary iodine equilibrium activity ratio = 20 to 1 (for 1 gpm leakage).

8) specific I-131 dose equivalent activity = 3.5 μCi/gm (Primary)
= 0.17 μCi/gm (Secondary).

9) gross specific activity in primary = $72/\overline{E}$ μCi/gm.

10) $X/Q = 7.0 \times 10^{-4}$ sec/m^3 at limiting point beyond site boundary of 1046 meters for 30 m release height - equivalent to ground level release due to topography including building wake effect for 5 percentile meteorology.

11) total gross radioactivity in primary coolant released to secondary coolant released to environs.

12) ten percent of the combined radioiodine activity from primary activity in secondary coolant and secondary activity present in steam mass (released to environs) assumed released to environs.

The whole body dose resulting from immersion in the cloud containing the released activity would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employed the simple model of the 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. The resulting whole body dose was determined to be less than 0.5 Rem for this accident. 2

The thyroid dose from the steam generator tube rupture accident has been analyzed assuming a tube rupture at full load and loss of offsite power at the time of the reactor trip, which results in steam release through the relief valves in the period before the faulty steam generator is isolated and primary system pressure is reduced. The limiting iodine activities for the primary and secondary systems are used in the initial conditions. One-tenth of the iodine contained in the liquid which is converted to steam and passed through the relief valves is assumed to reach the site boundary. The resulting thyroid dose from the combined primary and secondary iodine activity released to the environs was determined to be 1.5 Rem for this accident.

The limit for secondary iodine activity is consistent with the limits on primary system iodine activity and primary-to-secondary leakage of 1gpm. If the activity should exceed the specified limits following a power transient, the major concern would be whether additional fuel defects had developed bringing the total to above expected levels. From the observed removal of excess activity by decay and cleanup, it should be apparent whether activity is returning to a level below the specification limit. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the makeup tank gases to the waste gas decay tanks.

 Corrected pages, dtd. 5-23-75

3:3

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]).

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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.a 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.

3.1.6.3.b If the leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day (0.104 gpm), a reactor shutdown shall be initiated within 4 hours and the reactor shall be in the cold shutdown condition within the next 30 hours.

3.1.6.4 Deleted

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 10CFR20.

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, three 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 two other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once per shift; otherwise, be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.1.6.8 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which

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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 and 3.1.6.6 except that such losses when added to leakage shall not exceed 30 gpm.

3.1.6.9 If the reactor coolant system pressure isolation valve leakage is greater than the values given in Table 3.1.6.9, isolate (by having at least two valves in the high pressure piping closed*) the high pressure portion of the affected system from the low pressure portion within 4 hours and apply Specification 3.3.6, or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

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, the first indication 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 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.

*The motor operated valve shall remain closed and power supplies deenergized.

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.

The steam generator tube leakage limit (i.e., primary to secondary leakage limit) in Specification 3.1.6.3 is intended to assure timely shutdown of the plant for appropriate corrective action before rupture of the steam generator tubes occurs under normal operating or postulated accident conditions. These limits also serve to provide added assurance that the dosage contribution from tube leakage will be limited to a small fraction of 10CFR100 limits for a design basis steam generator tube rupture or main steam line break event.

<u>References</u>

FSAR Section 4.2.3.8

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TABLE 3.1.6.9

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[a][b][c] Allowable Leakage
Decay Heat Removal Train A	DH-14A	≤ 5.0 GPM
	DH-13A, DH-17	≤ 5.0 GPM (both valves together total)
Decay Heat Removal Train B	DH-14B	≤ 5.0 GPM
	DH-13B, DH-18	≤ 5.0 GPM (both valves together total)

Footnote:

[a] 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

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

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

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

[b] Minimum differential test pressure shall not be less than 150 psig.

[c] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

3.1.7 Moderator Temperature Coefficient of Reactivity Specification

3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is ≥95% of rated thermal power and shall be less positive than 0.9×10^{-4} $\Delta k/k/°F$ whenever thermal power is <95% of rated thermal power and the reactor is not shutdown.

3.1.7.2 The MTC shall be determined to be within its limits by confirmatory measurements prior to initial operation above 5% of rated thermal power after each fuel loading. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.

3.1.7.3 With the MTC outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

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 Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power, the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ $\Delta k/k/°F$ corrected to 95% of rated power. The most limiting event for positive MTC, the Startup Accident, has been analyzed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ $\Delta k/k/°F$.

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 to the right of the criticality limit of Figure 3.1.2-2. The shutdown margin shall be maintained greater than or equal to that specified in the COLR 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 vessel level instrument vent shall be checked for accumulation of undissolved gases. The temperature, pressure and dissolved gas concentration shall be restored to within their limits within 24 hours or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

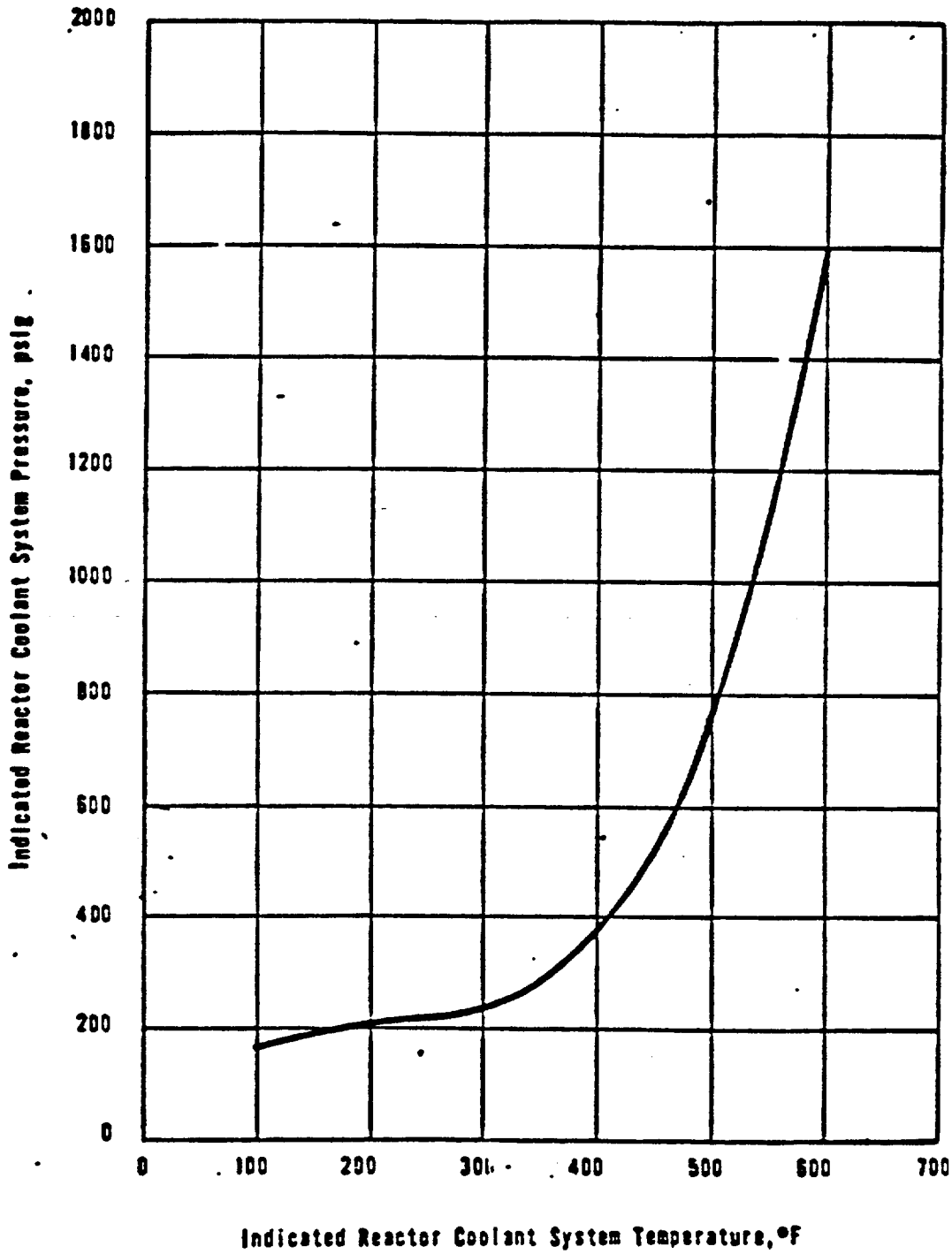
Bases

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 vessel level instrument vent 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 and one associated boric acid pump being operable. This tank shall contain at least the equivalent of the boric acid volume and concentration requirements of Figure 3.2-1 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 a temperature of at least 10°F above the crystallization temperature for the concentration in the tank.

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 (200°F) at the worst time in core life with a stuck control rod assembly and after xenon decay.

Minimum volumes (including a 10% safety factor) as specified by Figure 3.2-1 for the boric acid addition tank or an operable borated water storage tank (3) will each satisfy this requirement. The specification assures that adequate supplies are available whenever the reactor is heated above 200°F 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.

The alternate method of addition is to inject boric acid from the borated water storage tank using 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

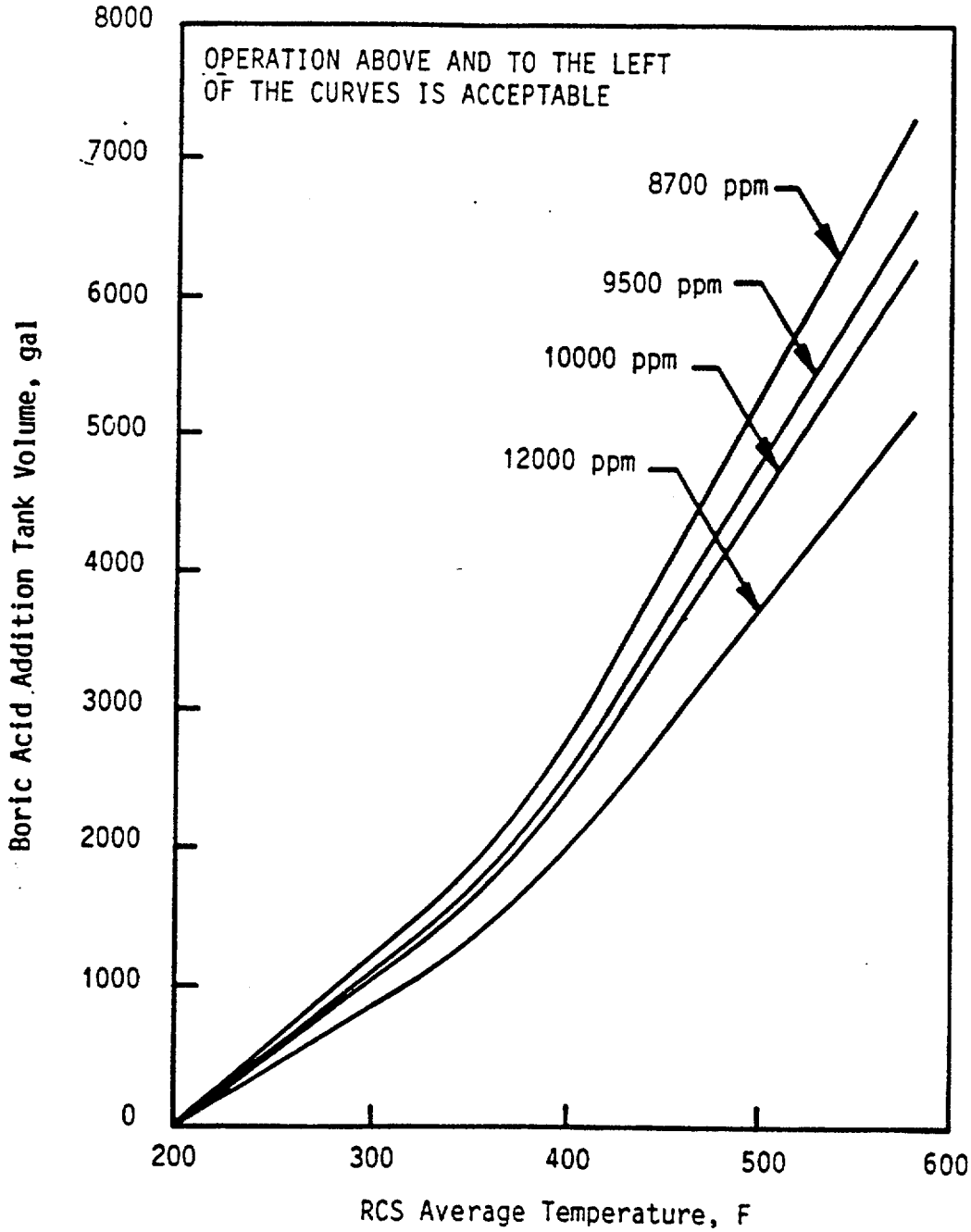
1. FSAR, Section 9.1; 9.2

2. FSAR, Figure 6-2

3. SAR, Section 3.1

Bases Change of 9-15-95

Boric Acid Addition Tank Volume and Concentration Vs RCS Average Temperature -- ANO-1, Cycle 8
Figure 3.2-1



Temp., F	Required Volume, gal			
	8700 ppm	9500 ppm	10000 ppm	12000 ppm
579	7308	6657	6306	5200
532	6126	5580	5289	4355
500	5273	4802	4548	3749
400	2790	2543	2409	1984
300	1234	1129	1070	877
200	0	0	0	0

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

Applicability

Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Objectivity

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

Specification

3.3.1 The 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) One train of reactor building emergency cooling.

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

(D) Two engineered safety feature actuated Low Pressure Injection (LPI) pumps shall be operable.

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

(F) Two Borated Water Storage Tank (BWST) level instrument channels shall be operable.

(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open.

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

(I) 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 350F 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 buses, 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 ± 0.4 feet (1040 ± 30 ft^3) of borated water at 600 ± 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 critical 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 two trains of reactor building emergency cooling. The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses.

(B) The sodium hydroxide tank shall contain a volume of ≥9,000 gallons of sodium hydroxide solution at a concentration >5.0 wt% and <16.5 wt%.

(C) All manual valves in the main discharge lines of the 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 Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water, reactor building spray and reactor building emergency cooling

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systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours, and, if not corrected, in cold shutdown condition within an additional 72 hours.

3.3.7 Exceptions to 3.3.6 shall be as follows:

(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.

(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure of level) shall be operable.

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within 7 days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

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(E) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore the inoperable reactor building emergency cooling train to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

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 emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train 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 emergency cooling and iodine removal. Specification 3.3.1 assures that the required equipment is operable.

A train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation.

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])

370,100 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. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal 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 minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core.

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 train of reactor building emergency cooling and spray train are operable.

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.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. 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. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

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 severance, limit the peak clad temperature to less than 2200°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.

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 6.3

(6) ANO Calculation 91-E-0019-01

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 decay heat by use of two steam generators.

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

3. A minimum of 11.1 feet (107,000 gallons) of water is available in Tank T41B.

4. (Deleted)

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

3.4.2 Components required to be operable by Specification 3.4.1 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.

3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

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* Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

3.4.4 If the conditions specified in 3.4.3 cannot be met:

1. With the motor driven EFW pump or its associated flow path inoperable and RCS conditions above CSD and RCS temperature < 280°F and any Steam Generator relied upon for heat removal, immediately initiate action to restore the EFW train to operable status.

2 With the RCS temperature ≥ 280°F and one steam generator supply path to the turbine driven EFW pump inoperable, restore the steam generator supply path to operable status within 7 days or be in Hot Shutdown within 6 hours and reduce RCS temperature to < 280°F within the next 12 hours.

3. With the RCS temperature ≥ 280°F and one EFW pump or its associated flow path inoperable, restore the EFW train to operable status within 72 hours or be in Hot Shutdown within 6 hours, and reduce RCS temperature to < 280°F within the next 12 hours.

4. With the RCS temperature ≥ 280°F, both EFW pumps or their associated flow paths inoperable, and the Auxiliary Feedwater pump available, be in Hot Shutdown within 6 hours, and reduce RCS temperature to < 280°F within the next 12 hours.

5. With the RCS temperature ≥ 280°F and both EFW pumps or their associated flow paths inoperable, and the Auxiliary Feedwater pump unavailable, immediately initiate action to restore one EFW train or the Auxiliary Feedwater pump to operable status. LCO 3.0.3 and all other LCO Required Actions requiring mode changes are suspended until one EFW train or the Auxiliary Feedwater pump is restored to operable status.

Bases

The Emergency Feedwater (EFW) system is designed to provide flow sufficient to remove heat load equal to 3½ percent full power operation. The system minimum flow requirement to the steam generator(s) is 500 gpm. This takes into account a single failure, pump recirculation flow, seal leakage and pump wear.

In the event of loss of main feedwater, feedwater is supplied by the emergency feedwater pumps, one which is powered from an operable emergency bus and one which is powered from an operable steam supply system. Both EFW pumps take suction from tank T41B. Decay heat is removed from a steam generator by steam relief through the turbine bypass, atmospheric dump valves, or safety valves. Fourteen of the steam safety valves will relieve the necessary amount of steam for rated reactor power.

The EFW system is considered to be operable when the components and flow paths required to provide EFW flow to the steam generators are operable. This requires that the turbine driven EFW pump be operable with redundant steam supplies from each of the main steam lines upstream of the MSIVs (CV-2617 and CV-2667) and capable of supplying EFW flow to either of the two steam generators. The motor driven EFW pump and associated flow path to the EFW system is also required to be operable. The piping, valves, instrumentation, and controls in the required flow paths shall also be operable. One EFW train, which includes the motor driven EFW pump, is required to be operable when above CSD and below 280°F with any steam generator relied upon for heat removal. This is because of reduced heat removal requirements, the short duration EFW would be required, and the insufficient steam supply available in this condition to power the turbine driven EFW pump.

When one of the required EFW trains is inoperable, action must be taken to restore the train to operable status within 72 hours. This condition includes loss of the steam supply to the turbine driven EFW pump. The 72 hour completion time is reasonable, based on the redundant capabilities afforded by the EFW system, time needed for repairs, and the low probability of a DBA occurring during this time period.

With two EFW trains inoperable, the unit must be placed in a mode in which the LCO does not apply using the Auxiliary Feedwater pump. With RCS temperature < 280°F the Decay Heat Removal system may be placed in operation.

With both EFW trains inoperable and the Auxiliary Feedwater pump unavailable, the unit is in a seriously degraded condition with only limited means for conducting a cooldown using nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW pump or the Auxiliary Feedwater pump to Operable status. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

The minimum amount of water in tank T41B 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 could be brought into operation within 4 hours or less.

A portion of tank T41B is protected from tornado missiles. The protected volume is sufficient to provide a thirty minute supply of water. This thirty minute period is sufficient to allow manual operator action, if required, to transfer suction of the emergency feedwater pumps to service water.

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 may be used to lock the channel trip relay in the untripped state as indicated by a light. Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6. Only one channel bypass key shall be accessible for use in the control room. While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

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

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.

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3.5.1.7 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig.

3.5.1.8 The degraded voltage monitoring relay settings shall be as follows:

a. The 4.16 KV emergency bus undervoltage relay setpoints shall be ≥1600 VAC but ≤3000 VAC with a time delay setpoint of ≥0.30 seconds and ≤0.98 seconds.

b. The 480 V emergency bus undervoltage relay setpoints shall be ≥423.2 VAC but ≤436.0 VAC with a time delay setpoint of 8 seconds ±1 second.

3.5.1.9 The following Reactor Trip circuitry shall be operable as indicated:

1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a and item 35 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.)

2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a and item 41 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 45% reactor power.)

3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.

3.5.1.10 Deleted

3.5.1.11 For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed.

3.5.1.12 The Containment High Range Radiation Monitoring instrumentation shall be operable with a minimum measurement range from 1 to 10^7 R/hr.

3.5.1.13 Two control room ventilation radiation monitoring channels shall be operable whenever the reactor coolant system is above the cold shutdown condition or during handling of irradiated fuel.

3.5.1.14 The Main Steam Line Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 10^{-1} to 10^{4} mR/hr, whenever the reactor is above the cold shutdown condition.

3.5.1.15 Initiate functions of the EFIC system which are bypassed at cold shutdown conditions shall have the following minimum operability conditions:

a. "low steam generator pressure" initiate shall be operable when the main steam pressure exceeds 750 psig.

b. "loss of 4 RC pumps" initiate shall be operable when neutron flux exceeds 10% power.

c. "main feedwater pumps tripped" initiate shall be operable when neutron flux exceeds 10% power.

3.5.1.16 The automatic steam generator isolation system within EFIC shall be operable when main steam pressure is greater than 750 psig.

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. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection system and other interconnected systems is evaluated. The affected reactor protection channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administrative controls are established to preclude placing a reactor protection channel in channel bypass when any other reactor protection channel contains an inoperable function in operation in the untripped state.

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.

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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 the RPS, the ESAS, and the EFIC enables 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 EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. Prior to placing a channel of EFIC in maintenance bypass, any NI/RPS channel containing inoperable functions in the untripped state is evaluated for its effect on EFIC. Only the EFIC channel corresponding to the NI/RPS channel containing the inoperable function may be placed in maintenance bypass unless it can be shown that the failure in the NI/RPS channel has no effect on EFIC actuation, actions are taken to ensure EFIC actuation when required, or the appropriate actions of Table 3.5.1-1 are implemented. The EFIC can be tested from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activation of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the component. The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.

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.

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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 4.16 KV bus undervoltage relay settings are based on a maximum setting, which is below the lowest allowed motor terminal momentary voltage of 75% of motor voltage rating of 4000 V. The settings are further reduced to include channel uncertainties and calibration tolerances. The 480V bus undervoltage relay settings are based on long term motor voltage requirements plus the maximum feeder voltage drop allowance resulting in a nominal 92% setting of motor rated voltage of 460 V. These settings are also adjusted to include channel uncertainties and calibration tolerances.

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

The subcooled margin monitors (SMM), and core-exit thermocouples (CET), Reactor Vessel Level Monitoring System (RVLMS) and Hot Leg Level Measurement System (HLLMS) are a result of the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737. The function of the ICC instrumentation is to increase the ability of the plant operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37 and are not required by the accident analysis, nor to bring the plant to cold shutdown conditions. The Reactor Vessel Level Monitor is provided as a means of indicating level in the reactor vessel during accident conditions. The channel operability of the RVLMS is defined as a minimum of three sensors in the upper plenum region and two sensors in the dome region operable. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced variables that may offset the sensor outputs. The channel operability of the HLLMS is defined as a minimum of one wide range and any two of the narrow range transmitters in the same channel operable. If the equipment is inaccessible due to health and industrial safety concerns (for example, high radiation area, low oxygen content of the containment atmosphere) or due to physical location of the fault (for example, probe failure in the reactor vessel), then operation may continue until the next scheduled refueling outage and a report filed.

The principal function of the Control Room Isolation-High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. Due to the unique arrangement of the shared control room envelope, one control room isolation channel receives a high radiation signal from the ANO-1 control room ventilation intake duct monitor and the redundant channel receives a high radiation signal from the ANO-2 control room ventilation intake duct monitor. With no channel of the control room radiation monitoring system operable, the CREVS must be placed in a condition that does not require the isolation to occur (i.e., one operable train of CREVS is placed in the emergency recirculation mode of operation). Reactor operation may continue indefinitely in this state.

To support loss of main feedwater analyses, steam line/feedwater line break analyses, SBLOCA analyses, and NUREG-0737 requirements, the EFIC system is designed to automatically initiate EFW when:

1. all four RC pumps are tripped
2. both main feedwater pumps are tripped
3. the level of either steam generator is low
4. either steam generator pressure is low
5. ESAS ECCS actuation (high RB pressure or low RCS pressure)

The EFIC system is also designed to isolate the affected steam generator on a steam line/feedwater line break and supply EFW to the intact generator according to the following logic:

- If both SG's are above 600 psig, supply EFW to both SG's.
- If one SG is below 600 psig, supply EFW to the other SG.
- If both SG's are below 600 psig, but the pressure difference between the two SG's exceeds 100 psig, supply EFW only to the SG with the higher pressure.
- If both SG's are below 600 psig and the pressure difference is less than 100 psig, supply EFW to both SG's.

At cold shutdown conditions all EFIC initiate and isolate functions are bypassed except low steam generator level initiate. The bypassed functions will be automatically reset at the values or plant conditions identified in Specification 3.5.1.15. "Loss of 4 RC pumps" initiate and "low steam generator pressure" initiate are the only shutdown bypasses to be manually initiated during cooldown. If reset is not done manually, they will automatically reset. Main feedwater pump trip bypass is automatically removed above 10% power.

REFERENCE

FSAR, Section 7.1

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

REACTOR PROTECTION 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 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 (Cont'd)

REACTOR PROTECTION SYSTEM (Cont'd)	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
11. Reactor trip upon loss of Main Feedwater	4	2	2	1	Notes 1, 15
12. Reactor trip upon turbine trip	4	2	2	1	Notes 1, 16
13. Electronic (SCR) Trip Relay	2	2	2	0	Note 23
14. Control Rod Drive Trip Breakers					
A. AC Breakers	2	2	2	0	Notes 24, 25
B. DC Breakers (Note 25)	2	2	2	0	Notes 24, 25

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 (Cont'd)

ENGINEERED SAFEGUARDS ACTUATION SYSTEM (Cont'd)

	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
EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM					
1. EFW Initiation					
a. Manual	2	1	2	1	Note 1

TABLE 3.5.1-1 (cont'd)

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM (Cont'd)

	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
b. Low Level SG A or B	4/SG	2/SG	2/SG	1	Note 1
c. Low Pressure, SG A or B	4/SG	2/SG	2/SG	1	Note 1, 19
d. Loss of Both MFW Pumps and PWR > 10%	4	2	2	1	Note 1
e. Loss of 4 RC Pumps	4	2	2	1	Note 1, 15
f. ESAS Actuation Logic Tripped	2	1	2	1	Note 1
2. SG-A Main Steam Line Isolation					
a. Manual	2	1	2	1	Note 1
b. Low SG A Pressure	4	2	2	1	Note 1, 19
3. SG-B Main Steam Line Isolation					
a. Manual	2	1	2	1	Note 1
b. Low SG B Pressure	4	2	2	1	Note 1, 19

TABLE 3.5.1-1 (cont'd)

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM (Cont'd)

	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
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. (cont'd)

OTHER SAFETY RELATED SYSTEMS
(Cont'd)

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 if conditions of column 3 or 4 cannot be met
2. Pressurizer level channels	2	N/A	2	1	Note 10
3. Emergency Feedwater flow channels	2/S.G.	N/A	1	0	Note 10
4. RCS subcooling margin monitors	2	N/A	1	0	Note 10
5. Electromatic relief valve flow monitor	2	N/A	1	0	Note 11
6. Electromatic relief block valve position indicator	1	N/A	1	0	Note 12
7. Pressurizer code safety valve flow monitors	2/valve	N/A	1/valve	0	Note 10
8. Degraded Voltage Monitoring					
a. 4.16 KV Emergency Bus Undervoltage	2/Bus	1/Bus	2/Bus	0	Note 14
b. 460 V Emergency Bus Undervoltage	*1/Bus	1/Bus	1/Bus	0	Notes 13, 14
9. Deleted					
10. Containment High Range Radiation Monitoring	2	N/A	2	0	Note 20
11. Containment Pressure - High Range	2	N/A	2	0	Note 21
12. Containment Water Level - Wide Range	2	N/A	2	0	Note 21

*Two undervoltage relays per bus are used with a coincident trip logic (2-out-of-2)

Table 3.5.1-1 (cont'd)

OTHER SAFETY RELATED SYSTEMS (Cont'd) 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 if conditions of column 3 or 4 cannot be met
13. In core Thermocouples (core-exit thermocouples)	6/core quadrant	N/A	2/core quadrant	0	Note 22
14. Control Room Radiation Monitors	2	1	2	1	Note 17, 18
15. Reactor Vessel Level Monitoring System	2	N/A	2	0	Note 28, 29
16. Hot Leg Level Measurement System (HLLMS)	2	N/A	2	0	Note 28, 29
17. Main Steam Line Radiation Monitors	1 /steam line	N/A	1/steam line	0	Note 30

TABLE 3.5.1-1 (Cont'd)

Notes:

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

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 four 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, the actions required by Column 5 shall apply.

7. These channels initiate control rod withdrawal inhibits not reactor trips at -10% rated power. Above 10% rated power, those 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.

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.

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TABLE 3.5.1-1 (Cont'd)

12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours.

13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies.

14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

15. This trip function may be bypassed at up to 10% reactor power.

16. This trip function may be bypassed at up to 45% reactor power.

17. With no channel operable, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig.

20. With one channel inoperable, (1) either restore the inoperable channel to operable status within 7 days, <u>or</u> (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

21. With one channel inoperable, restore the inoperable channel to operable status within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in HOT SHUTDOWN within 12 hours.

Table 3.5.1-1 (cont'd)

Notes:

22. With the number of operable channels less than two (2) per core quadrant restore the inoperable channel to operable status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

23. With the number of operable Electronic (SCR) Trip relays one less than the total number of Electronic (SCR) Trip relays in a channel, restore the inoperable Electronic (SCR) Trip relay to operable status in 48 hours or place the SCRs associated with the inoperable Electronic (SCR) Trip relay in trip in the next hour. With two or more Electronic (SCR) Trip relays inoperable, place all Electronic (SCR) Trip relays associated with that channel in trip in the next hour. This requirement does not apply to the Electronic Trip channels associated with Group 8 Regulating Power Supply.

24. With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

    a. Within 1 hour:

        1. Place the inoperable channel in the tripped condition, or

        2. Remove power supplied to the control rod trip device associated with the inoperable channel.

    b. One additional channel may be bypassed for up to 4 hours for surveillance testing and the operable channel above may be bypassed for up to 30 minutes in any 24-hour period when necessary to test the trip breaker associated with the logic of the channel being tested. The inoperable channel above shall not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.

25. With one of the Control Rod Drive Trip Breaker diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

26. Interrupts motor power to the Safety Groups of control rods only.

27. Deleted

Table 3.5.1-1 (cont'd)

28. With the number of OPERABLE channels one less than the minimum number of channels required to be OPERABLE:

    a. If repairs are feasible, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

    b. If repair is not feasible without shutting down, operation may continue and a special report pursuant to specification 6.12.5 shall be submitted to the NRC within 30 days following the failure; describing the action taken, the cause of the inoperability, and the plans and schedules for restoring the channel to OPERABLE status during the next scheduled refueling outage.

29. With the number of OPERABLE channels two less than the minimum channels required to be OPERABLE:

    a. If repairs are feasible, restore at least one inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

    b. If repair is not feasible without shutting down, operation may continue and a special report pursuant to specification 6.12.5 shall be submitted to the NRC within 30 days following the failure; describing the action taken, the cause of the inoperability, and the plans and schedules for restoring the channels to OPERABLE status during the next scheduled refueling outage.

30. With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and: 1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or 2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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 greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn. With the shutdown margin less than that required, immediately initiate and continue boration injection until the required shutdown margin is restored.

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 existence of an available shutdown margin greater than or equal to that specified in the COLR. 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 an available shutdown margin greater than or equal to that specified in the COLR 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 contained 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 single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Power Tilt:

1. Except for physics tests, if quadrant power tilt exceeds the tilt limit set in the CORE OPERATING LIMITS REPORT, reduce power so as not to exceed the allowable power level for the existing reactor coolant pump combination less at least 2% for each 1% tilt in excess of the tilt limit.

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

   a. The Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

   b. The control rod group and APSR withdrawal limits shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

   c. The reactor power imbalance setpoints shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

3. If quadrant power 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 power 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 20% ± 5 between two sequential groups, except for physics tests.

3. Except for physics tests or exercising control rods, the control rod position setpoints are specified in the CORE OPERATING LIMITS REPORT for 4, 3, AND 2 pump operation. If the applicable control rod position setpoints are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 4 hours.

4. Except for physics tests or exercising axial power shaping rods (APSRs), the limits for APSR position are specified in the CORE OPERATING LIMITS REPORT.

   With the APSRs outside the specified limit provided in the CORE OPERATING LIMITS REPORT, corrective measures shall be taken immediately to achieve the correct position. Acceptable APSR positions shall be attained within 4 hours.

3.5.2.6 Reactor Power Imbalance:

1. Reactor power imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power.

2. Except for physics tests, reactor power imbalance shall be maintained within the envelope defined by the CORE OPERATING LIMITS REPORT.

3. If the reactor power imbalance is not within the envelope defined by the CORE OPERATING LIMITS REPORT, corrective measures shall be taken to achieve an acceptable reactor power imbalance.

4. If an acceptable reactor power imbalance is not achieved within 4 hours, reactor power shall be reduced until reactor power imbalance setpoints 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.

<u>Bases</u>

The reactor power-imbalance envelope defined in the CORE OPERATING LIMITS REPORT is based on either LOCA analyses (which have defined the maximum linear heat rate (see CORE OPERATING LIMITS REPORT), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria) or loss of forced reactor coolant flow analysis (such that the hot fuel rod does not experience a departure from nucleate boiling condition). Corrective measures will be taken immediately should the indicated quadrant power tilt, control rod position, or reactor power imbalance be outside their specified boundaries. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA or loss of forced reactor coolant flow occur is highly improbable because all of the power distribution parameters (quadrant power tilt, rod position, and reactor power imbalance) must be at their limits while

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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.
e. Fuel rod bowing.

The 20 ±5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower parts of the stroke. Control rods are arranged in groups or banks defined as follows:

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

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full-out position ([1]). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% Δk/k at rated power. These values have been shown to be safe by the safety analysis ([2]) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% Δk/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% Δk/k at beginning of life, hot zero power, would result in a lower transient peak thermal power and therefore less severe environmental consequences than a 0.65% Δk/k ejected rod worth at rated power.

Control rod Groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 20%. The normal position at power is for Groups 6 and 7 to be partially inserted.

*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.

The quadrant power tilt limits set forth in the CORE OPERATING LIMITS REPORT 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 setpoints in the CORE OPERATING LIMITS REPORT, ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

The quadrant power tilt limits and reactor power imbalance setpoints in the CORE OPERATING LIMITS REPORT, apply when using the plant computer to monitor the limits. The 2-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service. Additional uncertainty is applied to the limits when other monitoring methods are used.

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

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	≤30 psig (44.7 psia)
	High Pressure Injection	≤ 4 psig (18.7 psia)
	Start of Reactor Building Cooling and Reactor Building Isolation	≤ 4 psig (18.7 psia)
	Reactor Bldg. Ventilation	≤ 4 psig (18.7 psia)
	Low Pressure Injection	≤ 4 psig (18.7 psia)
	Penetration Room Ventilation	≤ 4 psig (18.7) psia)
Low Reactor Coolant System Pressure**	High Pressure Injection	≥ 1585 psig
	Low Pressure Injection	≥ 1585 psig
	Start of Reactor Building Cooling and Reactor Building Isolation	≥ 1585 psig

*May be bypassed during reactor building leak rate test.

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

With the safety features actuation setpoints less conservative than the above values, declare the channel inoperable and apply the applicable Action requirements of Table 3.5.1-1 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the trip setpoint value.

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 1585 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation. (1)

REFERENCES

(1) FSAR, Section 14.2.2.5

(2) B&W Calculation 32-1158581

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 three strings shall lie in the same axial plane with one 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 four detectors shall lie in each core half. Each set of four 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.

With the incore detector system inoperable, do not use the system for the above applicable monitoring function. The provisions of Specification 3.0.3 are not applicable.

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 startup, 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.

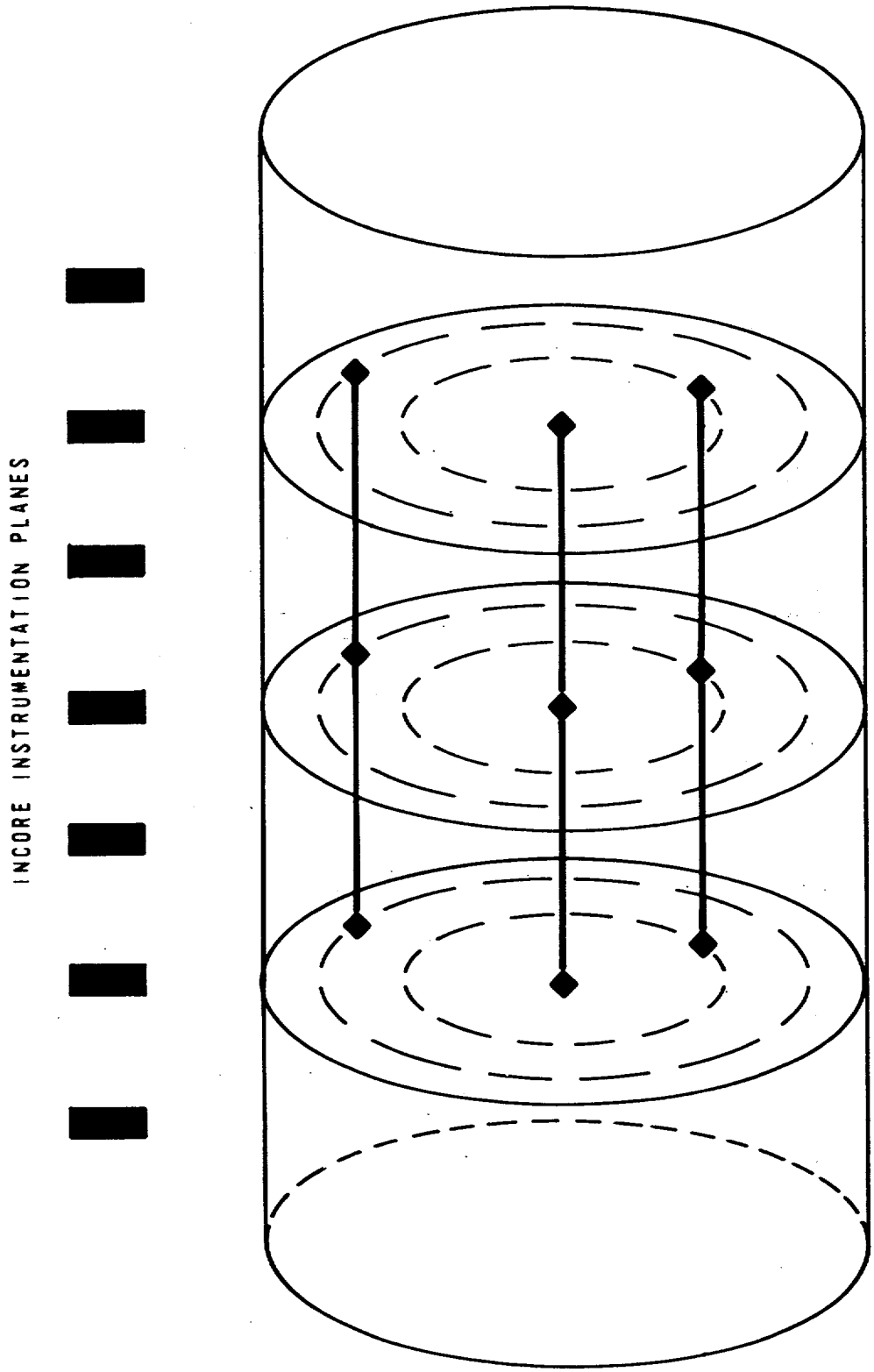
3. Figure 3.5.4-3 combines Figures 3.5.4-1 and 3.5.4-2 to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions. Startup testing will verify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

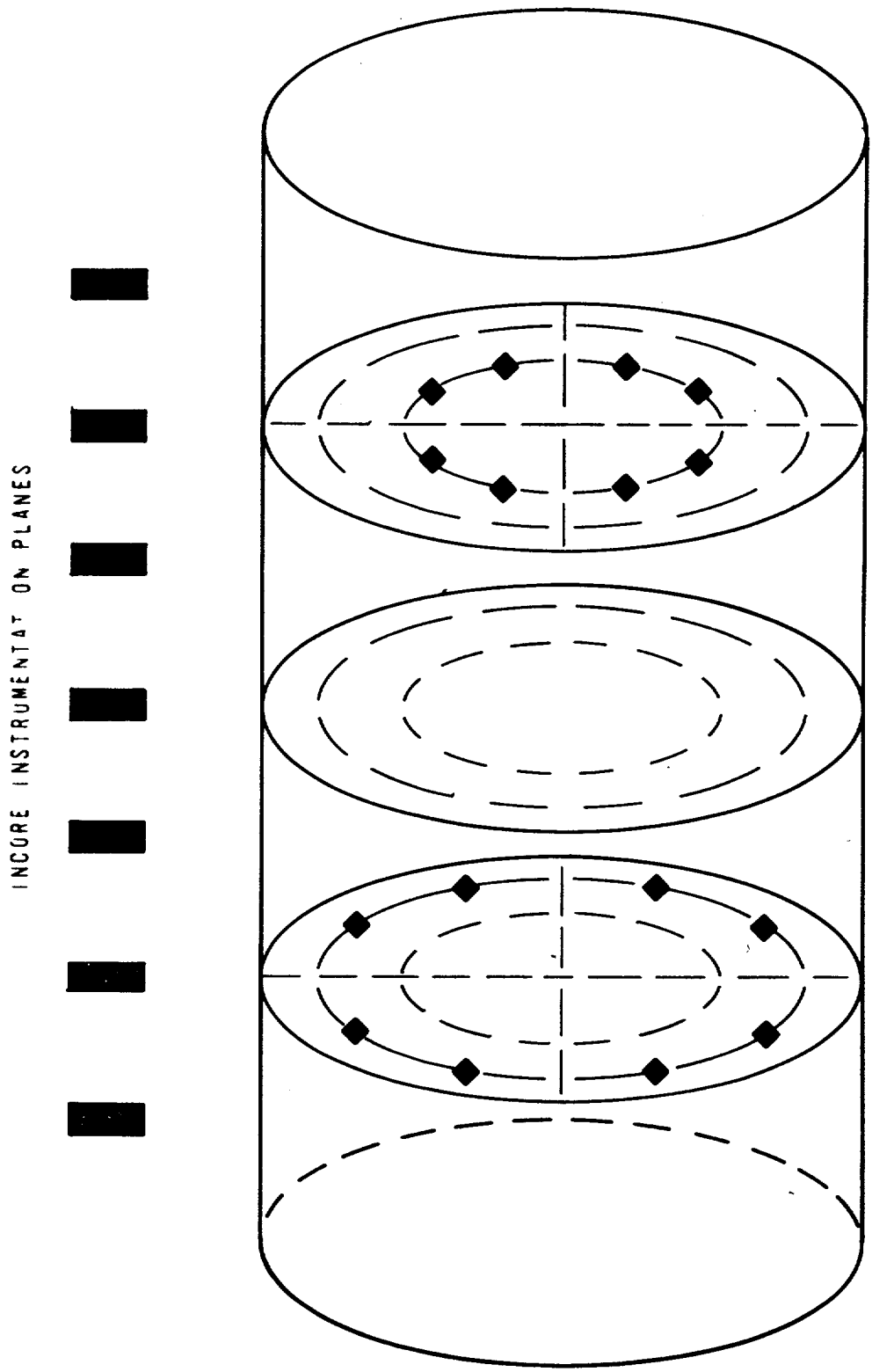
detectors in specified locations is not operable, power will be decreased to or below 80 percent for the operating reactor coolant pump combination.

REFERENCE

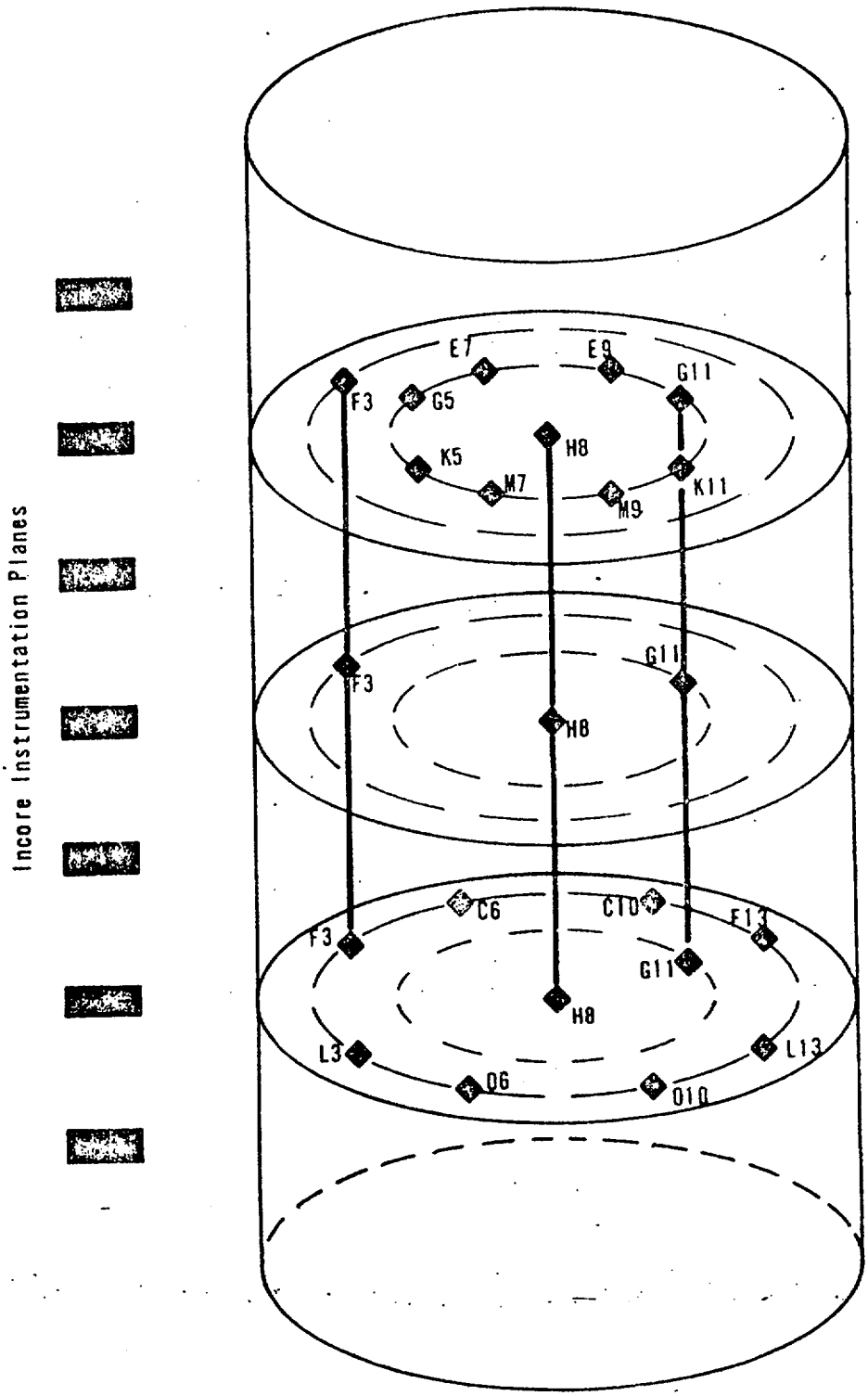
(1) FSAR, Section 4.1.1.3



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

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3.6 REACTOR BUILDING

Applicability

Applies to the operability of the reactor building.

Objective

To assure reactor building operability.

Specification

3.6.1 The reactor building shall be operable 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.

With the reactor building inoperable, restore the reactor building to operable status within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

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. The provisions of Specification 3.0.3 are not applicable.

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. The provisions of Specification 3.0.3 are not applicable.

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. With the reactor critical, restore the containment pressure to within its limits within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

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. The provisions of Specification 3.0.3 are not applicable.

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

Included in reactor building operability are both the reactor building integrity as defined in Specification 1.7 and the reactor building structural integrity. Structural integrity limitations as described in the ANO Containment Inspection Program ensure the reactor building will be maintained comparable to the original design standards throughout the facility life span. Visual and other required examinations of tendons, anchorages and surfaces are performed periodically in accordance with station procedures. These procedures embody applicable requirements of the 1992 Edition with the 1992 Addenda of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code as set forth in 10 CFR 50.55a(g)(6)(ii)(B). Any degradations exceeding the Containment Inspection Program acceptance criteria during inspection surveillances will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall containment operability, if any.

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.

REFERENCE

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, 480 V motor control centers and 120 V AC distribution panels 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 each with:

1. a separate day tank containing a minimum of 160 gallons of fuel,

2. a separate emergency storage tank containing a minimum of 138 inches (20,000 gallons) of fuel,

3. a separate fuel transfer pump, and

4. a separate starting air compressor.

D. DELETED

E. DELETED

F. The off-site power undervoltage and protective relaying interlocks associated with required startup transformer power sources shall be operable per Table 3.5.1-1.

G. The selective load-shed features associated with Startup Transformer No. 2 shall be operable if selected for auto transfer.

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 200F. 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. Startup Transformer No. 2 may be removed from service for up to 30 days as part of a preplanned preventative maintenance schedule. The 30-day allowance may be applied not more than once in any 10-year period. The provisions of Specification 3.0.4 are not applicable to Startup Transformer No. 2 during the 30-day preventative maintenance period.

C. Either one of the two diesel generators may be inoperable for up to 7 days in any month provided that 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. The operability of the remaining diesel generator shall be demonstrated within 24 hours unless it is determined that a common cause failure does not exist or, unless it is currently in operation or has been demonstrated operable within the previous 24 hours.

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. DELETED

F. DELETED

G. DELETED

H. If the requirements of Specification 3.7.1.G cannot be met, either:

(1) place all Startup Transformer No. 2 feeder breakers in "pull-to-lock" within 1 hour, restore the inoperable interlocks to operable status within 30 days, or submit within 30 days a Special Report pursuant to Specification 6.12.5 outlining the cause of the failure, proposed corrective action and schedule for implementation; or

(2) apply the action requirements of Table 3.5.1-1, Note 14.

3.7.3 Both 125 VDC electrical power subsystems shall be operable when the unit is above the cold shutdown condition.

A. With one 125 VDC electrical power subsystem inoperable:

1. verify that there are no inoperable safety related components associated with the operable 125 VDC electrical subsystem which are redundant to the inoperable 125 VDC electrical power subsystem,

2. verify the operability of the diesel generator associated with the operable 125 VDC electrical subsystem immediately, and

3. restore the 125 VDC electrical subsystem to operable status within 8 hours.

B. With one 125 VDC electrical power subsystem inoperable, and unable to satisfy the requirements or allowable outage times of 3.7.3.A.1, 3.7.3.A.2, or 3.7.3.A.3, the unit shall be placed in hot shutdown within 12 hours and in cold shutdown within an additional 24 hours.

3.7.4 Battery cell parameters shall be within limits when the associated 125 VDC electrical power subsystems are required to be operable.

A. With one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category A or B limits:

1. Within 1 hour, verify pilot cell electrolyte level and float voltage meet Table 4.6-1 Category C limits,

2. Within 24 hours and once per 7 days thereafter, verify battery cell parameters meet Table 4.6-1 Category C limits, and

3. Within 31 days, restore battery cell parameters to Table 4.6-1 Category A and B limits.

B. With one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category A or B limits and unable to satisfy the requirements or allowable outage times of 3.7.4.A.1, 3.7.4.A.2, or 3.7.4.A.3, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

C. With one or more batteries with electrolyte temperature of the pilot cell not within the limits of Specification 4.6.2.8, electrolyte temperature of representative cells not within the limits of Specification 4.6.2.6 or with one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category C limits, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

Bases

The electrical system is designed to be electrically self-sufficient and provide adequate, reliable power sources for all electrical equipment during startup, normal operation, safe shutdown and handling of all emergency situations. To prevent the concurrent loss of all auxiliary power, the various sources of power are independent of and isolated from each other.

In the event that the offsite power sources specified in 3.7.1.A (1 or 2) are inoperable, the required capacity of one emergency storage tank plus one day tank (20, 160 gallons) will be sufficient for not less than three and one-half days operation for one diesel generator loaded to full capacity. (ANO-1 FSAR 8.2.2.3) The underground emergency storage tanks are gravity fed from the bulk storage tank and are normally full, while the day tanks are fed from transfer pumps which are capable of being cross connected at their suction and discharges and automatically receive fuel oil when their inventory is less than 180 gallons. Thus, at least a seven day total diesel oil inventory is available onsite for emergency diesel generator operation during complete loss of electric power conditions.

Technical Specification 3.7.2 allows for the temporary modification of the specifications in 3.7.1 provided that backup system(s) are operable with safe reactor operation and combined availability of the engineered safety features ensured. The requirements for diesel generators are consistent with Generic Letter 84-15, "Proposed Staff Actions to Improve And Maintain Diesel Generator Reliability" and the Revised Standard Technical Specifications (NUREG 1430). The evaluation of a common cause failure (degradation that may affect the operability of the remaining diesel generator) should be completed within 24 hours from when the affected diesel generator is determined to be inoperable.

Technical Specifications 3.7.1.F and 3.7.1.G provide assurance that the Startup Transformer No. 2 loads will not contribute to a sustained degraded grid voltage situation. This will protect ESF equipment from damage caused by sustained undervoltage.

The 125 VDC electrical power system consists of two independent and redundant safety related class 1E DC electrical subsystems. Each subsystem consists of one 100% capacity 125 VDC battery, its associated battery charger, and its distribution network. Additionally, there is one spare battery charger per subsystem, which provides backup service in the event that the preferred battery charger is out of service.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, no operable battery charger, or inoperable battery and no operable associated battery charger), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power subsystems with attendant loss of ES functions, continued power operation should not exceed 8 hours.

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational event or a postulated design basis accident. Cell parameter limits are conservatively established, allowing continued DC electrical system function even with Table 4.6-1 Category A and B limits not met.

With one or more cells in one or more batteries not within limits (i.e., Table 4.6-1 Category A limits not met, or Category B limits not met, or Category A and B limits not met) but within the Table 4.6-1 Category C limits, the battery is degraded but has sufficient capacity to perform its intended function. Therefore, the battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period of time. The pilot cell electrolyte level and float voltage are required to be verified to meet the Table 4.6-1 Category C limits within 1 hour (TS 3.7.4.A.1). These checks will provide a quick representative status of the remainder of the battery cells. Verification that the Table 4.6-1 Category C limits are met (TS 3.7.4.A.2) provides assurance that

during the time needed to restore the parameters to within the Category A and B limits, the battery will still be capable of performing its intended function. This verification is repeated at 7 day intervals until the parameters are restored to within Category A and B limits. This periodic verification is consistent with the increased potential to exceed these battery parameter limits during these conditions.

With one or more batteries with one or more battery cell parameters outside the Table 4.6-1 Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured. Therefore, the battery must be immediately declared inoperable and the corresponding DC electrical power subsystem must be declared inoperable.

Additionally, other potentially extreme conditions, such as electrolyte temperature of the pilot cell falling below 60°F, average electrolyte temperature of representative cells falling below 60°F or battery terminal voltage below the limit are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

An allowance has been provided, extending the allowable outage time for Startup Transformer No. 2 only, for up to 30 days. The 30-day allowance is permitted not more than once in any 10-year period, which is considered sufficient for proper maintenance of the transformer. The 30-day window should permit extensive preplanned preventative maintenance without placing either unit in an action statement of short duration and would allow both units to be operating during such maintenance. Because this allowance assumes parts are prestaged, appropriate personnel are available, and proper contingencies have been established, it is not intended to be used for an unexpected loss of the transformer. Pre-established contingencies will consider the projected stability of the offsite electrical grid, the atmospheric stability projected for the maintenance window, the ability to adequately control other ongoing plant maintenance activities that coincide with the window, projected flood levels, and the availability of all other power sources. Since a station blackout is the most affected event that could occur when power sources are inoperable, the steam driven emergency feedwater pump will also be maintained available during the evolution.

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.a. At least one decay heat removal loop shall be in operation.* Otherwise, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system, and close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours.

b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable.**

Otherwise, immediately initiate corrective action to return the required loops to operable status as soon as possible.

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.

*The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of core alterations.

**The normal or emergency power source may be inoperable for each shutdown cooling loop.

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 capable* of being closed. The equipment hatch cover shall also be capable* of being closed. At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

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. The provisions of Specification 3.0.3 are not applicable.

3.8.10 The reactor building purge isolation system, including the radiation monitors shall be tested and verified to be operable within 7 days prior to refueling operations. The provisions of Specification 3.0.3 are not applicable.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 100 hours. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.

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 operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specification 3.0.3 are not applicable.

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. The provisions of Specification 3.0.3 are not applicable.

3.8.14 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specification 3.0.3 are not applicable.

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* Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed.

3.8.15 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable.

3.8.16 Storage in Region 2 (as shown on Figure 3.8.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable.

3.8.17 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.

3.8.18 During the handling of irradiated fuel, the control room emergency air conditioning system and the control room emergency ventilation system shall be operable as required by Specification 3.9.

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 9.6 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 requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.([1])

The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.([2]) Although the refueling boron concentration is sufficient to maintain the core keff ≤ 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 keff 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.

Per specification 3.8.6, the reactor building personnel and/or emergency airlock doors and the equipment hatch may be open during movement of irradiated fuel in the reactor building provided at least one door of each airlock and the equipment hatch are capable of being closed in the event of a fuel handling accident and the plant is in REFUELING SHUTDOWN with 23 feet of water above the fuel seated within the reactor pressure vessel. Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface.

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 100 hours ([3]); and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates.

Specifications 3.8.15 and 3.8.16 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.17 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

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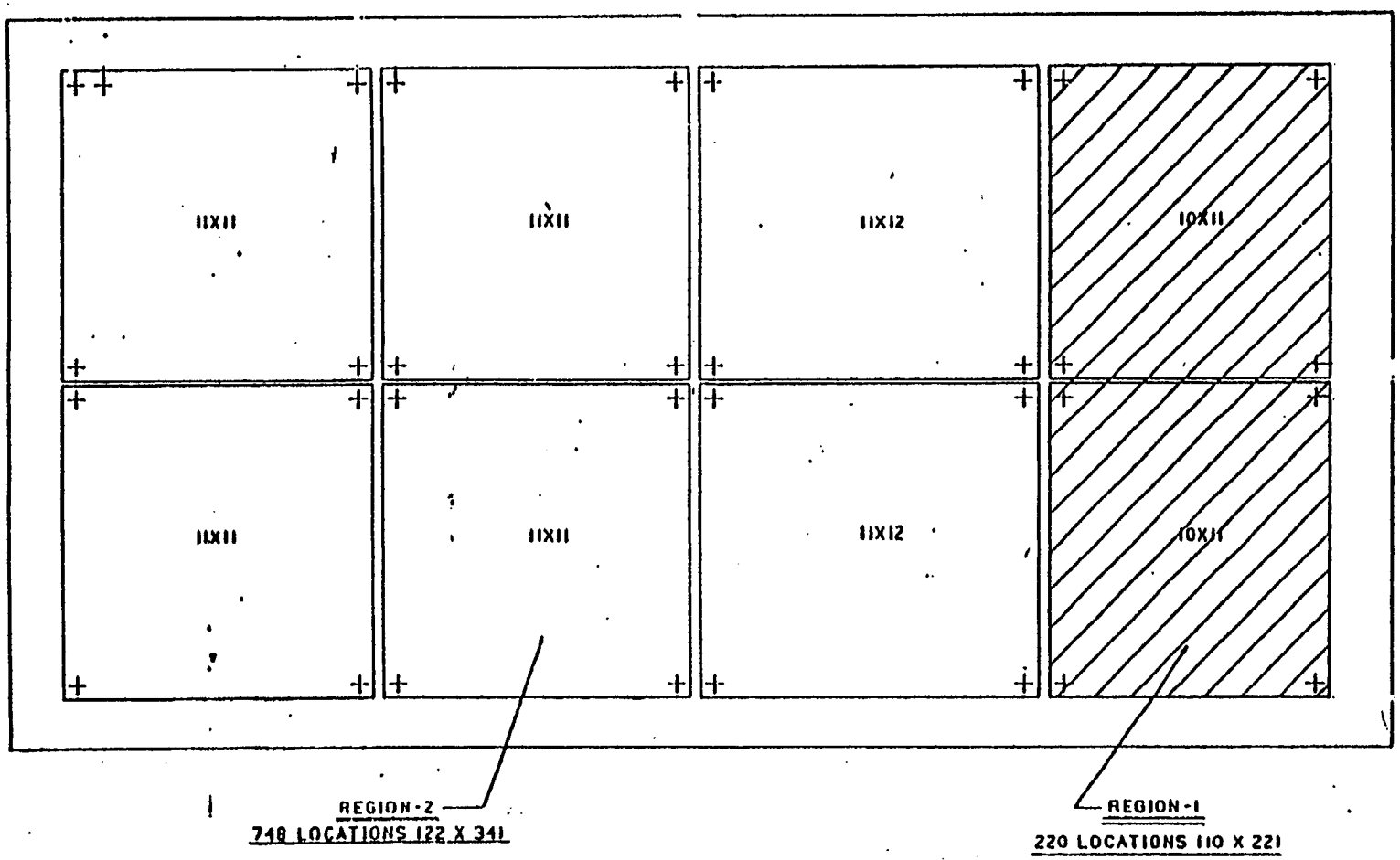


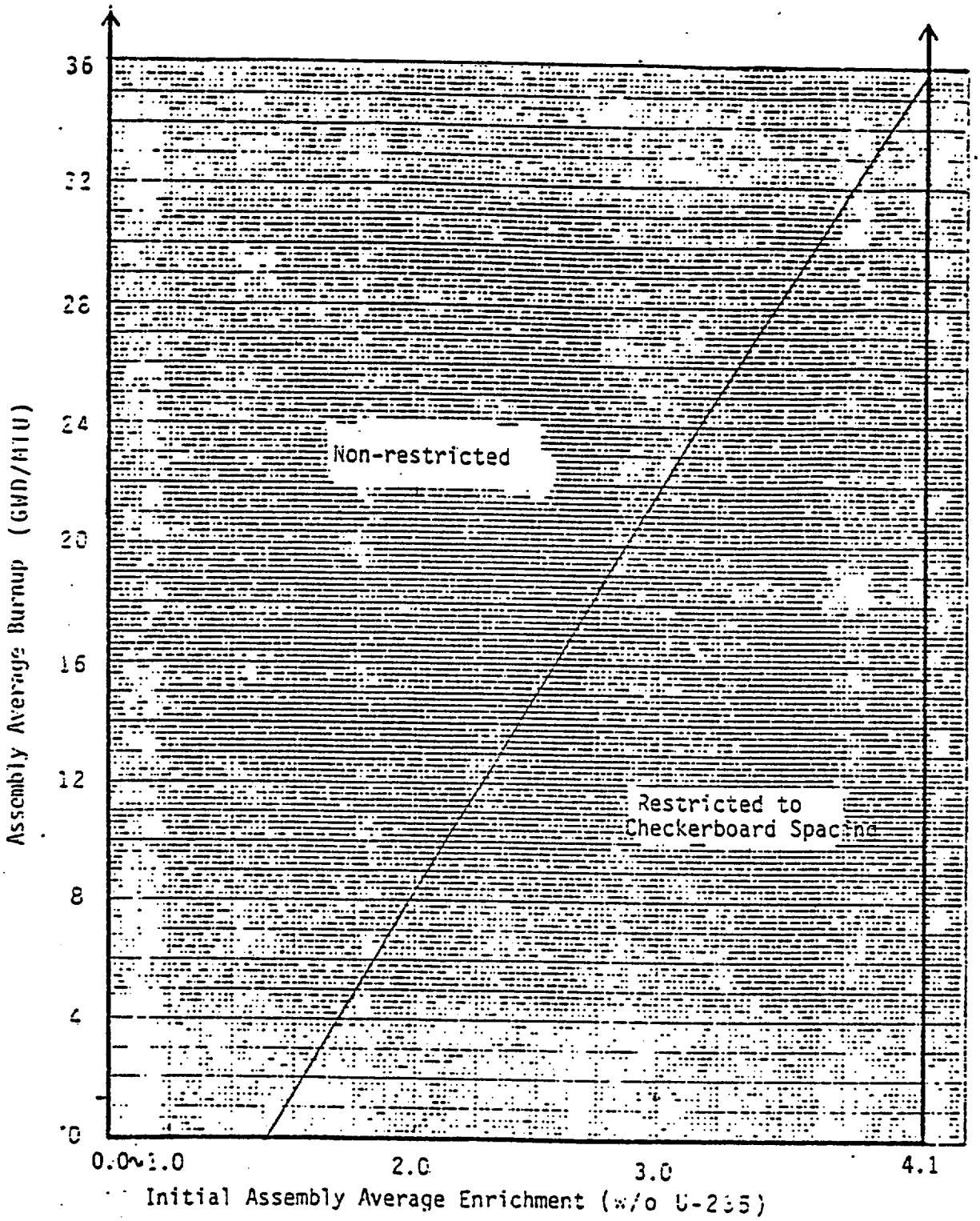
SPENT FUEL POOL ARRANGEMENT UNIT #1

Figure 3.8.1

FIGURE 3.8.2

MINIMUM BURNUP VS. INITIAL ENRICHMENT

FOR REGION 2 ENRICHMENT



3.9 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEMS

Applicability

Applies to the operability of the control room emergency ventilation and air conditioning systems.

Objective

To ensure that the control room emergency ventilation and air conditioning systems will perform within acceptable levels of efficiency and reliability.

Specification

3.9.1 Control Room Emergency Air Conditioning System

3.9.1.1 Two independent trains of the control room emergency air conditioning system shall be operable whenever the reactor coolant system is above the cold shutdown condition or during handling of irradiated fuel.

3.9.1.2 With one train of control room emergency air conditioning inoperable, restore the inoperable train to Operable status within 30 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.9.2 Control Room Emergency Ventilation System

3.9.2.1 Two independent trains of the control room emergency ventilation system shall be operable whenever the reactor coolant system is above the cold shutdown condition or during handling of irradiated fuel.

3.9.2.2 With one train of control room emergency ventilation inoperable, restore the inoperable train to Operable status within 7 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 30 hours.

Bases

The control room emergency ventilation and air conditioning system is designed to isolate the combined control rooms to ensure that the control rooms will remain habitable for Operations personnel during and following all credible accident conditions and to ensure that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system. The design configuration of the system is based on limiting the radiation exposure to personnel occupying the control room to 5 REM or less whole body, or its equivalent, in accordance with the requirements of General Design Criteria 19 of Appendix A, 10 CFR 50.

Unit 1 and Unit 2 control rooms are a single environment for emergency ventilation and air conditioning concerns. Since the control room emergency ventilation and air conditioning equipment is shared between units, the plant status of both units must be considered when determining applicability of the specification.

Due to the unique situation of the shared emergency ventilation and air conditioning equipment, the components may be cross fed from the opposite unit per predetermined contingency actions/procedures. Unit 1 may take credit for operability of these systems when configured to achieve separation and independence regardless of normal power and/or service water configuration. This will be in accordance with pre-determined contingency actions/procedures.

The control room emergency ventilation system consists of two independent filter and fan trains, two independent actuation channels and the Control Room isolation dampers. The control room dampers isolate the control room within 10 seconds of receipt of a high radiation signal.

If the actuation signal can not start the emergency ventilation recirculation fan, operating the affected fan in the manual recirculation mode and isolating the control room isolation dampers provides the required design function of the control room emergency ventilation system to isolate the combined control rooms to ensure that the control rooms will remain habitable for operations personnel during and following accident conditions. This contingency action should be put in place immediately (within 1 hour) to fully satisfy the design functions of the control room emergency ventilation system.

The control room emergency air conditioning system (CREACS) provides temperature control for the control room following isolation of the control room. It is manually started from the Unit 2 Control Room. The CREACS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment.

With both trains of the control room emergency ventilation and/or emergency air conditioning inoperable, the function of the control room emergency air systems have been lost, requiring immediate action to place the reactor in a condition where the specification does not apply.

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 I-131 dose equivalent of the radioiodine activity in the secondary coolant shall not exceed 0.17 uCi/gm. With the secondary coolant activity in excess of 0.17 uCi/gm I-131, be in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours.

Bases

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside containment and a loss of load incident were considered.

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 or steam line break, there are only small quantities of these gases which would be released.

The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for Specification 3.1.4.1 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 uCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for Specification 3.1.4.1. For the less probable accident of a steam line break, the assumption is made that a loss of 1×10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 uCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident.

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 emergency cooling pond shall be operable whenever containment integrity is established as required by Specification 3.6.1 with:

1. A minimum contained water volume of 70 acre-feet (equivalent to an indicated water level of 5 feet).

2. An average water temperature of ≤ 100°F.

3.11.2 With the requirements of Specification 3.11.1 not satisfied, be in the hot shutdown condition within 6 hours and in the cold shutdown condition within the following 30 hours.

Bases

The requirements of Specification 3.11.1 provide for sufficient water inventory in the emergency cooling pond to mitigate within acceptable limits the effects of a DBA with a concurrent failure of the Dardanelle Reservoir. The minimum water depth takes into account (1) water loss from evaporation due to heat load and climatological conditions, (2) pond bottom irregularities, (3) suction pipe level at the pond and (4) operator action in transferring the service water system from the Dardanelle Reservoir. Operator action is credited in the inventory analysis during the transfer of the service water system to the pond. Specifically, pump returns are transferred to the pond shortly after a loss of lake event and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the pond, lake water is pumped into the pond, increasing level. This additional water is required, along with that maintained by Technical Specifications, to ensure a 64.5 inch pond depth, which corresponds to a 30 day supply of cooling water.

The values are based on worst case initial conditions which could be present considering a simultaneous normal shutdown of Unit 1 and emergency shutdown of Unit 2 following a LOCA in Unit 2, using the ECP as a heat sink. The measured ECP temperature at the discharge from the pond is considered a conservative average of total pond conditions since solar gain, wind speed, and thermal current effects throughout the pond will essentially be at equilibrium conditions under initial stagnant conditions.

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 5 μCi or less of alpha emitting material. The provisions of Specification 3.0.3 are not applicable.

3.12.2 A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.12.5 within 90 days if source leakage tests reveal the presence of ≥0.005 microcuries of removable contamination.

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

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3.13 PENETRATION ROOM VENTILATION SYSTEM

Applicability

Applies to the operability of the penetration room ventilation system.

Objective

To ensure that the penetration room ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.13.1 Two independent circuits of the penetration room ventilation system shall be operable whenever reactor building integrity is required with the following performance capabilities:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flow (± 10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis from the charcoal adsorber banks shall show the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

c. Fans shall be shown to operate within ± 10% of design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (± 10%).

e. Air distribution shall be uniform within ± 20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system.

f. Each circuit of the system shall be capable of automatic initiation.

3.13.2 If one circuit of the penetration room ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days provided that during such seven days all active components of the other circuit shall be operable.

3.13.3 If the requirements of Specifications 3.13.1 and 3.13.2 cannot be met, the reactor shall be placed in the cold shutdown condition within 36 hours.

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Bases

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

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should ensure a radioactive methyl iodide removal efficiency of a least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If one circuit of the penetration room ventilation system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue for a limited period of time while repairs are being made.

3.14 HYDROGEN RECOMBINERS

Applicability

Applies to the operating status of the hydrogen recombiner systems.

Objective

To ensure that the hydrogen recombiner systems will perform within acceptable levels of efficiency and reliability.

Specification

3.14.1 Two independent hydrogen recombiner systems shall be operable whenever reactor building integrity is required.

3.14.2 With one hydrogen recombiner system inoperable, restore the inoperable system to operable status within 30 days or the reactor shall be placed in the hot shutdown condition within the next 6 hours.

3.14.3 Hydrogen concentration instruments shall be operable.

3.14.4 With one of two hydrogen concentration instruments inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least hot shutdown within the next 6 hours.

Bases

The hydrogen recombiner systems are designed to operate as necessary to limit the hydrogen concentration in the reactor building following a Loss of Coolant Accident.

The system is composed of two redundant 100% capacity Internal Electrical Hydrogen Recombiners, manufactured by Westinghouse.

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3.15 FUEL HANDLING AREA VENTILATION SYSTEM

Applicability

Applies to the operability of the fuel handling area ventilation system.

Objective

To ensure that the fuel handling area ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.15.1 The fuel handling area ventilation system shall be in operation whenever irradiated fuel handling operations are in progress in the fuel handling area of the auxiliary building and shall have the following performance capabilities:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows (±10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

c. Fans shall be shown to operate within ±10% design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%).

e. Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system.

3.15.2 If the requirements of Specification 3.15.1 cannot be met, irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed). The provisions of Specification 3.0.3 are not applicable.

Bases

The fuel handling area ventilation system is designed to filter the auxiliary building atmosphere during fuel handling operations to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing two exhaust fans and a filter train. The fans are redundant and only one is required to be operating. The filter train consists of a prefilter, a HEPA filter and a charcoal adsorber in series.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine absorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should ensure a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

3.16 Shock Suppressors (Snubbers)

Applicability

This technical specification applies to all shock suppressors (snubbers). The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Objective

To assure adequate shock suppression protection for primary coolant system piping and any other safety related system or component under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. This is done by assuring the operability of those shock suppressors installed for that purpose.

Specification

3.16.1 With one or more applicable snubbers inoperable, within 72 hours either:

a. Replace or restore the inoperable snubbers to an OPERABLE status and perform an engineering evaluation of the attached components per Specification 4.16.1.f or,

b. Perform a review and evaluation which justifies continued operation with the inoperable snubber(s) and perform an engineering evaluation of the attached component(s) per Specification 4.16.1.f or,

c. Declare the attached system inoperable and follow the appropriate ACTION statement for that system.

Bases

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

Because the shock suppressor protection is required only during low probability events, a period of 72 hours is allowed for repairs, replacements or evaluations. If a reveiw and evaluation of an INOPERABLE snubber is preformed and documented to justify continued operation, and provided all design criteria are met with the INOPERABLE snubber, then the INOPERABLE snubber would not need to be restored or replaced. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures.

TABLE 3.16-1

SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS)

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TABLE 3.16-2

SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS)

(DELETED)

3.17 Fire Suppression Water System

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3.18 FIRE SUPPRESSION SPRINKLER SYSTEM

DELETED

Administrative Letter dt. 11-20-9-

3.19 CONTROL ROOM AND AUXILIARY CONTROL ROOM HALON SYSTEMS

DELETED

Annotation initial. DR 7-30-92

3.20 FIRE HOSE STATIONS

DELETED

Revision Letter dt 7-30-02

3.21 FIRE BARRIERS

DELETED

Correction letter dt 4-30-92

3.22 REACTOR BUILDING PURGE FILTRATION SYSTEM

Applicability

This specification applies to the operability of the reactor building purge filtration system.

Objective

To assure that the reactor building purge filtration system will perform within acceptable levels of efficiency and reliability.

Specification

3.22.1 The reactor building purge filtration system shall be operable whenever irradiated fuel handling operations are in progress in the reactor building and shall have the following performance capabilities:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows (±10%) on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity within ±20% of system design, 0.05 to 0.15 mg/m^3 inlet methyl iodide concentration, ≥70% R. H. and ≥125F.

c. Fans shall be shown to operate within ±10% design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%).

e. Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the reactor building purge filtration system.

3.22.2 If the requirements of Specification 3.22.1 cannot be met, either:

a. Irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed); or,

b. Isolate the reactor building purge system.

3.22.3 The provisions of Specification 3.0.3 are not applicable.

Bases

The reactor building purge filtration system is designed to filter the reactor building atmosphere during normal operations for ease of personnel entry into the reactor building. This specification is intended to require the system operable during fuel handling operations, if the system

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is to be used, to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing a supply and an exhaust fan and a filter train. The filter train consists of a pre-filter, a HEPA filter and a charcoal adsorber in series.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of a least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

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3.23 REACTOR BUILDING PURGE VALVES

APPLICABILITY

This specification applies to the reactor building purge supply and exhaust isolation valves.

OBJECTIVE

To specify that reactor building isolation purge valves be closed whenever containment integrity is required by TS 3.6.1.

SPECIFICATION

3.23.1 The reactor building purge supply and exhaust isolation valves shall be closed and handswitch keys removed whenever containment integrity is required by TS 3.6.1.

BASES

The reactor building supply and exhaust isolation valves are required to be closed during normal plant operation in order to ensure reactor building integrity. Purging is allowed only when containment integrity is not required by TS 3.6.1.

3.24 EXPLOSIVE GAS MIXTURE

Applicability

Applies to the Waste Gas System hydrogen/oxygen analyzers.

Objective

To prevent accumulation of explosive mixture in the waste gas system.

Specification

3.24.1 The concentration of hydrogen/oxygen shall be limited in the waste gas decay tanks to Region "A" of Figure 3.24-1.

3.24.2 When the hydrogen/oxygen concentration in any of the decay tanks enters Region "B" of Figure 3.24-1, corrective action shall be taken to return the concentration values to Region "A" within 24 hours.

3.24.3 The provisions of Specification 3.0.3 are not applicable.

Bases

These hydrogen/oxygen limits provide reasonable assurance that no hydrogen/oxygen explosion could occur to allow rupture of the waste gas decay tanks. The hydrogen and oxygen limits are based on information in NUREG/CR-2726 "Light Water Reactor Hydrogen Manual".

3.25 RADIOACTIVE EFFLUENTS

3.25.1 Radioactive Liquid Holdup Tanks

Applicability: At all times.

Objective: To ensure that the limits of 10 CFR 20 are not exceeded.

Specifications:

3.25.1 A. The quantity of radioactive material contained in each unprotected* outside temporary radioactive liquid storage tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

B. With the quantity of radioactive material exceeding the above limit, immediately suspend all additions of radioactive material to the affected tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.12.2.6.

C. The provisions of Specification 3.0.3 are not applicable.

Bases:

This specification is provided to ensure that in the event of an uncontrolled release of the contents of the tank* the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the unrestricted area.

*Tanks included in this specification are those outdoor temporary tanks that 1) are not surrounded by liners, dikes, or walls capable of holding the tank contents, and 2) do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

3.25.2 Radioactive Gas Storage Tanks

Applicability: At all times

Objective: To restrict the amount of activity in a radioactive gas holdup tank.

Specifications:

3.25.2 A. The quantity of radioactivity contained in each gas storage tank shall be limited to 78,782 curies noble gases (Xe-133 equivalent).

B. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.12.2.6.

C. The provisions of Specification 3.0.3 are not applicable.

Bases:

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that, in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a member of the public at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the operational modes or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the Action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The time at which the Action is taken may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the Action requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an operational mode or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to operational modes as required to comply with Action requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1,2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i),

4.0.5 (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

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.

4.1 (Continued)

b. Equipment and sampling test shall be performed as detailed in Table 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

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

4.0.2 Establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance intervals.

4.0.3 Establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the Action requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the Action requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the Action requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2 was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the Action requirements are less than 24 hours or a shutdown is required to comply with Action requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the Action requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with Action requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance

includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of mode changes imposed by Action requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the Action requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the Action requirements are applicable at the time that the surveillance is terminated. If the Action requirements are greater than 24 hours, sufficient time exists to complete the surveillance.

Surveillance Requirements do not have to be performed on inoperable equipment because the Action requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

4.0.4 Establishes the requirement that all applicable surveillances must be met before entry into an operational mode or other condition of operation specified in the Specification. The purpose of this Specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a mode or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in operational modes or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with Action requirements, the provision of Specification 4.0.4 do not apply because this would delay placing the facility in a lower mode of operation.

4.0.5 Establishes the requirement that inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

4.1 Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear 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 nonsteady 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 once every 18 months.

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 every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC 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.

All reactor protective channels will be tested before startup if the individual channel rotational frequency has been discontinued or if outage activities could potentially have affected the operability of one or more channels. A rotation will then be established to test the first Channel one week after startup, the second Channel two weeks after startup, the third Channel three weeks after startup, and the fourth Channel four weeks after startup.

The established reactor protective system instrumentation and EFIC 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 quarter. 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 4.1-3 are 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	Q	NA	
2. Control Rod Drive Trip Breaker	NA	Q(1)	NA	(1) To include independent testing of the shunt and undervoltage trip attachments.
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 core 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 System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, 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 other than containment high range monitors (item 57)				(1) Check functioning of self-checking feature on each detector.
	a. Process Monitoring System	S	Q	R	
	b. Area Monitoring System	S	M(1)	R	
	c. Main Steam Line Radiation Monitors	S	M	R	

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Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	
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) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.
31. Deleted				
32. Diesel generator protective relaying starting interlocks and circuitry	M	Q	NA	
33. Off-site power undervoltage and protective relaying interlocks and circuitry	W	R(1)	R(1)	(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2.
34. Borated water storage tank level indicator	W	NA	R	
35. Reactor trip upon loss of main feedwater circuitry	M	PC	R	

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
36. Boric Acid Addition Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
37. Degraded Voltage Monitoring	W	R	R	
38. Sodium Hydroxide Tank Level Indicator	NA	NA	R	
39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning
40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check
41. Reactor Trip Upon Turbine Trip Circuitry	M	PC	R	
42. Deleted				

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
43. ESAS Manual Trip Functions				
a. Switches & Logic	NA	R	NA	
b. Logic	NA	M	NA	
44. Reactor Manual Trip	NA	P	NA	
45. Reactor Building Sump Level	NA	NA	R	
46. EFW Flow Indication	M	NA	R	

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
47. RCS Subcooling Margin Monitor	D	NA	R	
48. Electromatic Relief Valve Flow Monitor	D	NA	R	
49. Electromatic Relief Block Valve Position Indicator	D	NA	R	
50. Pressurizer Safety Valve Flow Monitor	D	NA	R	
51. Pressurizer Water Level Indicator	D	NA	R	
52. Deleted				
53. EFW Initiation				
a. Manual	NA	M	NA	
b. SG Low Level, SGA or B	S	M	R	
c. Low Pressure SGA or B	S	M	R	
d. Loss of both MFW Pumps and PWR > 10%	S	M	R	

TABLE 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
e. Loss of 4 RC pumps	S	M	NA	
f. ESAS automatic logic tripped	NA	M	NA	
54. SGA main steam line isolation				
a. Manual	NA	M	NA	
b. SGA pressure low	S	M	R	
55. SGB main steam line isolation				
a. Manual	NA	M	NA	
b. SGB pressure low	S	M	R	
56. EFW valve commands (Vector)				
a. SG A pressure low	S	M	R	
b. SG B pressure low	S	M	R	
c. SG pressure difference SG A pressure> SG B pressure	S	M	R	
SG B pressure> SG A pressure	S	M	R	

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A High Range Level High-high	S	M	R	
e. SG B High Range Level High-high	S	M	R	
57. Containment High Range Radiation Monitors	D	M	R	
58. Containment Pressure-High	M	NA	R	
59. Containment Water Level-Wide Range	M	NA	R	
60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	
61. Core-exit Thermocouples	M	NA	R	
62 Electronic (SCR) Trip Relays	NA	Q	NA	
63 RVLMS	M	NA	R	
64 HLLMS	M	NA	R	

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable
SA - SA Twice per Year

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 Valve Every 18 Months
4.	Main Steam Safety Valves	Setpoint	Four Valves Every 18 Months
5.	Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown
6a.	Reactor Coolant System Leakage	Evaluate	Daily
b.	Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2
7.	Emergency-powered Pressurizer Heaters	Power availability	Daily
		Heater capacity functional test	Every 18 Months
8.	Reactor Building Isolation Trip	Functioning	Every 18 months
9.	Service Water Systems	Functioning	Every 18 months
10.	Spent Fuel Cooling System	Functioning	Every 18 months when irradiated fuel is in the pool

1/ Same as tests listed in Section 4.7

Notes:

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

Item	Test	Frequency
11. Decay Heat Removal System Isolation Valve Automatic Closure and Isolation System	Functioning	Each Refueling Shutdown
12. Flow Limiting Annulus on Main Feedwater Line 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.
13. Main Steam Isolation Valves	a. Exercise through approximately 10% travel	a. Quarterly
	b. Cycle	b. Every 18 months
14. Main Feedwater Isolation Valves	a. Exercise through approximately 5% travel	a. Quarterly
	b. Cycle	b. Every 18 months
15. Reactor Internals Vent Valves	Demonstrate operability by:	Each refueling shutdown.
	a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities.	
	b. Verifying that the valve is not stuck in an open position, and	
	c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).	

TABLE 4.1-2 (Continued)

Minimum Equipment Test Frequency

	Item	Test	Frequency
16.	RCS Vent Paths	Demonstrate operability by flow verification	At least once per 18 months during cold shutdown
17.	PORV	Exercise	End of each refueling outage

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Table 4.1-3
MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(7)
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)
	d. Dissolved Gases	d. Weekly (7)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)
	f. Boron Concentration	f. 3 times/week
	g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of $\overline{E}$. A radiochemical analysis and calculation of $\overline{E}$ and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes.

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity.

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2.

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above.

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition.

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition.

(9) Required only when fuel is in the pool and prior to transferring fuel to the pool.

(10) Not required when not generating steam in the steam generators.

(11) The following shall be required until the end of Cycle 2 operation:

a. Gross radioiodine shall be determined at least three times per week during power operation.

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b. If the steady state gross radioiodine concentration increases by a factor of ten or more, the NRC shall be promptly notified with a written followup per Specification 6.12.3.1.