DRAFT

APR 1 1985

TABLE OF CONTENTS

DRAFT

APR 1 1985

TABLE OF CONTENTS

DRAFT

APR 1 1985

TABLE OF CONTENTS

DRA

APR 1

TABLE OF CONTENTS

DRAFT

APR 1 1985

TABLE OF CONTENTS

DRAFT

APR 1 1985

TABLE OF CONTENTS

DRAFT
APR 1 1985

TABLE OF CONTENTS

DRAFT

APR 1 1985

SECTION 1.0

INTRODUCTION

AND

DEFINITIONS

DRAFT

APR 1 1985

INTRODUCTION

These Technical Specifications apply to the Fort St. Vrain Nuclear Generating Station Unit No. 1. These Technical Specifications pertain to certain features, characteristics and conditions governing the operation of this facility which are important in protecting the barriers in the facility that separate the radioactive materials in the facility from the environs.

These Technical Specifications will not be changed except by express permission and with the approval of the Nuclear Regulatory Commission.

DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output.

BASES (BASIS)

1.3 The BASES shall be that part of a Specification that summarizes the reasons for the Limiting Condition of Operation and for the Surveillance Requirements, but in accordance with 10 CFR 50.36 are not a part of the Technical Specifications.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and with the required accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable, considering systems design, to verify OPERABILITY including alarm, interlock and/or trip functions.

DRAFT

APR 1 1985

COMPARISON REGION

1.7 A COMPARISON REGION shall be a core refueling region whose power, flow, and coolant outlet temperature characteristics are used to determine the individual refueling region outlet temperature of a region for which the measured outlet temperature is unreliable. Experience has shown that Regions 20 and 32 through 37 have the potential for significant discrepancies between measured and actual region outlet temperature. These discrepancies are caused by a transverse flow of relatively cool helium from the core reflector interface along the region outlet thermocouple sleeve. This flow passes over the region outlet thermocouple assemblies of these regions and depresses the indicated outlet temperature.

CONGESTED CABLE AREAS(S)

1.8 CONGESTED CABLE AREA(S) shall be the THREE ROOM CONTROL COMPLEX, the area containing redundant cable concentrations on the Reactor Building side of the "J" wall, and the area containing redundant cable concentrations on the Turbine Building side of the "G" wall.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the PCRV that alters the core reactivity, while fuel is in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position or condition.

CORE AVERAGE TEMPERATURE

1.10 a. During SHUTDOWN and REFUELING, CORE AVERAGE TEMPERATURE shall be the arithmetic average of the CORE AVERAGE INLET TEMPERATURE and the CORE AVERAGE OUTLET TEMPERATURE.

b. During STARTUP, LOW POWER, and POWER OPERATION, CORE AVERAGE TEMPERATURE shall be a calculated value based on average core inlet and outlet temperature, primary coolant flow adjusted for bypass flow, and reactor power.

CORE AVERAGE INLET TEMPERATURE

1.11 The CORE AVERAGE INLET TEMPERATURE shall be the arithmetic average of the operating circulator inlet temperatures, with appropriate corrections for circulator power input, steam generator regenerative heat load, and PCRV liner cooling system heat loss.

CORE AVERAGE OUTLET TEMPERATURE

1.12 The CORE AVERAGE OUTLET TEMPERATURE shall be the arithmetic average of the INDIVIDUAL REFUELING REGION OUTLET TEMPERATURES.

DOSE EQUIVALENT I-131

1.13 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III if TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in Table E-7 of Nuclear Regulatory Commission Regulatory Guide 1.109, Revision 1, October 1977.

E-bar - AVERAGE DISINTIGRATION ENERGY

1.14 E-bar shall be the average (weighted in proportion to the concentration of each noble gas radionuclide in the sample) of the sum of the average beta and gamma energies per disintigration (MeV/d) for the noble gas radionuclides in the sample.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Specification 4.0.3, or as otherwise specified in the Surveillance Requirement.

INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE

1.16 The INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE shall be defined as follows:

a. For Regions 1 through 19 and 21 through 31, the measured refueling region outlet temperature.

b. For Regions 20 and 32 through 37, whichever of the following temperatures is hottest: 1) the measured refueling region outlet temperature or 2) the refueling region outlet temperature based upon the following quantities:

1. The ratio of the relative power in each of these regions to that in their COMPARISON REGIONS as determined from physics calculations.

2. The ratio of the helium flow rate through each of these regions to that through their COMPARISON REGIONS as determined based upon inlet orifice valve positions.

3. The measured refueling region outlet temperatures of their COMPARISON REGION.

IRRADIATED FUEL

1.17 IRRADIATED FUEL shall be fuel that has a radiation level greater than or equal to 100 mr/hr measured perpendicularly one foot from a fuel element surface.

LIMITING SAFETY SYSTEM SETTING(S)

1.18 LIMITING SAFETY SYSTEM SETTING(S) shall be the trip setpoint specified in Specification 2.2, for the automatic protective devices that ensure corrective action to prevent exceeding the SAFETY LIMITS.

MEMBER(S) OF THE PUBLIC

1.19 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 licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OPERABLE - OPERABILITY

1.20 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, 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).

OPERATING - IN OPERATION

1.21 A system, subsystem, train, component or device shall be OPERATING or IN OPERATION when it is actually performing its specified function(s).

DRAFT

APR 1 1985

OPERATIONAL MODE- MODE

1.22 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of Reactor Mode Switch Setting, Interlock Sequence Switch Setting, and % RATED THERMAL POWER, specified in Table 1.0-1.

PHYSICS TESTS

1.23 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 13 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANT PROTECTIVE SYSTEM (PPS)

1.24 The PLANT PROTECTIVE SYSTEM (PPS) shall be the reactor protective circuitry and the circuitry that protects various plant components from major damage. This system initiates (1) scram, (2) loop shutdown, (3) circulator trip, and (4) rod withdraw prohibit functions as specified in Specification 3/4.3.1.

POWER-TO-FLOW RATIO (P/F)

1.25 POWER-TO-FLOW RATIO (P/F) shall be the percentage of RATED THERMAL POWER divided by the percentage of Design PRIMARY COOLANT FLOW at RATED THERMAL POWER.

PRIMARY COOLANT FLOW

1.26 The PRIMARY COOLANT FLOW shall be the sum of the helium massflow (lb/hr) for each of the operating circulators. The Design PRIMARY COOLANT FLOW at RATED THERMAL POWER is 3.5E+06 lb/hr.

REACTOR PRESSURES

1.27 REACTOR PRESSURES shall be:

a. NORMAL WORKING PRESSURE (NWP) = 688 psig.

b. PEAK WORKING PRESSURE (PWP) = 704 psig. PWP includes allowance for transients and variations in PCRV helium pressure control.

c. REFERENCE PRESSURE (RP) = 845 psig. RP is the maximum internal pressure allowed over the PCRV 30-year operating life except for the initial pressure test.

d. NORMAL OPERATING PRESSURE (NOP). NOP varies in the POWER OPERATION range between 610 psia and 700 psia.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be the maximum design reactor core heat generated by nuclear fission, equal to 842 MWt.

REFUELING CYCLE

1.29 REFUELING CYCLE shall be that interval of time between successive scheduled refuelings of a significant (greater than or equal to one-tenth) portion of the core.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions that require NRC notification or written report under the requirements of 10 CFR 10.72 and 10 CFR 50.73.

SAFE SHUTDOWN COOLING

1.31 SAFE SHUTDOWN COOLING shall be the removal of core stored energy and decay heat using Safe Shutdown Equipment (See FSAR Table 1.4-2). The reactivity condition in the core during SAFE SHUTDOWN COOLING shall be subcritical.

SAFETY LIMIT

1.32 SAFETY LIMIT(S) shall be limitations on process variables as identified in Specification 2.1. These limitations are defined to protect the fuel particle integrity and the integrity of the primary coolant system boundaries.

SHUTDOWN MARGIN

1.33 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming that all OPERABLE control rod pairs are fully inserted except for the single control rod pair of highest reactivity worth which is capable of being withdrawn.

SITE BOUNDARY

1.34 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

DRAFT

APR 1 1985

STAGGERED TEST BASIS

1.35 A STAGGERED TEST BASIS shall consist of:

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

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

SURVEILLANCE INTERVAL

1.36 The SURVEILLANCE INTERVAL shall be the period of time between performance of Surveillance Requirements. Specific intervals are identified in Specification 4.0.3.

THERMAL POWER

1.37 THERMAL POWER shall be the total reactor core heat generated by nuclear fission, as determined by an appropriate heat balance calculation, or from calibrated nuclear instrumentation.

THREE ROOM CONTROL COMPLEX

1.38 The THREE ROOM CONTROL COMPLEX shall be that area of the turbine building which includes the Control Room, the Auxiliary Electric Room, and the 480 Volt Switchgear Room.

TRIP

1.39 TRIP shall be the switching of an instrument or a device with two stable states from its normal state to its abnormal state. The result of a TRIP on a system level may be control rod scram, pressure relief, loop shutdown, etc.

UNRESTRICTED AREA

1.40 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

TABLE 1.0-1

OPERATIONAL MODES

MODE	INTERLOCK SEQUENCE SWITCH SETTING	REACTOR MODE SWITCH SETTING	% RATED THERMAL POWER*
POWER OPERATION (P)	Power	Run	Greater Than 30%
LOW POWER (L)	Low Power	Run	Greater Than 5% Less Than or Equal to 30%
STARTUP (S/U)	Startup	Run	Less Than or Equal to 5%
SHUTDOWN (S/D)	**	Off	0
REFUELING (R)***	**	Fuel Loading	0

* Excluding decay heat.
** Interlock Sequence Switch (ISS) may be in any position in SHUTDOWN and REFUELING
*** Includes Reactor Internal Maintenance, See Specification 3/4.9.1.

DRAFT

APR 1 1985

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

DRAFT

APR 1 1985

2.1 SAFETY LIMITS

REACTOR CORE SAFETY LIMIT

2.1.1 The total Integrated Operating Time and times of individual transients with core POWER-TO-FLOW RATIOS above the curve of Figure 2.1.1-2 at the appropriate power level during the lifetime of any fuel segment shall not exceed the following limits:

2.1.1.1 POWER-TO-FLOW RATIOS Less Than or Equal to 1.17:

a. Determination of Transient Operating Time: For each transient resulting in POWER-TO-FLOW RATIOS exceeding the curve of Figure 2.1.1-2, but less than or equal to 1.17, the Transient Operating Time shall be that time interval during which the curve of Figure 2.1.1-2 has been exceeded.

b. Limits: The Transient Operating Time for each individual transient in this range shall be limited such that the total Integrated Operating Time for all such transients shall not exceed 100 hours.

2.1.1.2 POWER-TO-FLOW RATIO Greater Than 1.17 and Less than or Equal to 2.5:

a. Determination of Transient Operating Time: For each transient resulting in POWER-TO-FLOW RATIOS above 1.17 and less than or equal to 2.5, the Transient Operating Time shall be that time interval during which the curve of Figure 2.1.1-2 has been exceeded, not including the first 120 seconds.

b. Limits: The combination of the reactor core POWER-TO-FLOW RATIO and the total Integrated Operating Time at this POWER-TO-FLOW RATIO during the lifetime of any segment shall not exceed the SAFETY LIMIT given in Figure 2.1.1-1. This SAFETY LIMIT is exceeded when the combination of operating parameters (power, flow, and time) lies above or to the right of the line given in Figure 2.1.1-1.

2.1.1.3 POWER-TO-FLOW RATIOS Greater than 2.5 and Less Than or Equal to 15:

a. Determination of Transient Operating Time: For each transient resulting in POWER-TO-FLOW RATIOS above 2.5 and less than or equal to 15, the Transient Operating Time shall be that time interval from the point where the POWER-TO-FLOW RATIO exceeds the limit of Figure 2.1.1-2, until it drops below 2.5, not including the first 100 seconds.

DRAFT

APR 1 1985

2.1.1 REACTOR CORE SAFETY LIMIT (Continued)

b. Limits: The Transient Operating Time for each individual transient in this range shall be limited such that the total Integrated Operating Time with POWER-TO-FLOW RATIOS above 2.5 and less than or equal to 15 shall not exceed 2 minutes. The allowable additional times for POWER-TO-FLOW RATIOS less than 2.5 are given in Specification 2.1.1.2.

2.1.1.4 POWER-TO-FLOW RATIOS Greater Than 15:

a. Determination of Transient Operating Time: For each transient resulting in POWER-TO-FLOW RATIOS above 15, the Transient Operating Time shall be that time interval from the point where the POWER-TO-FLOW RATIO exceeds the limit of Figure 2.1.1-2, until it drops below 2.5, not including the first 60 seconds.

b. Limits: The Transient Operating Time for each individual transient in this range shall be limited so that the total Integrated Operating Time with POWER-TO-FLOW RATIOS above 15 shall not exceed 2 minutes. The allowable additional times for POWER-TO-FLOW RATIOS less than 2.5 are given in Specification 2.1.1.2.

APPLICABILITY: POWER OPERATION and LOW POWER MODES ABOVE*

ACTION:

a. Individual Transients With POWER-TO-FLOW RATIOS Less Than or Equal to 1.17

During an individual transient, if the combination of POWER-TO-FLOW RATIO and percentage of RATED THERMAL POWER exceeds the curve of Figure 2.1.1-2, but does not exceed a POWER-TO-FLOW RATIO of 1.17, restore the plant to below the curve of Figure 2.1.1-2 within 30 minutes, or be in STARTUP within 12 hours and be in SHUTDOWN within the next 12 hours, and comply with the requirements of Specification 6.7.

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* Applicable only above 15% of RATED THERMAL POWER.

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2.1.1 REACTOR CORE SAFETY LIMIT (Continued) APR 1 1985

b. Individual Transients With POWER-TO-FLOW RATIOS Above 1.17

1) For transients which result in POWER-TO-FLOW RATIOS between 1.17 and 2.5, immediately reduce power to lower the POWER-TO-FLOW RATIO to less than 1.17. If this corrective action is not successful within two minutes, an immediate shutdown shall be initiated, be in SHUTDOWN within one hour, and comply with the requirements of Specification 6.7.

2) If an individual Transient Operating Time limit of Specification 2.1.1.3 or 2.1.1.4 is exceeded, an immediate shutdown shall be initiated, be in SHUTDOWN within one hour of initiation of the transient, and comply with the requirement of Specification 6.7.

c. Integrated Operating Time Limits

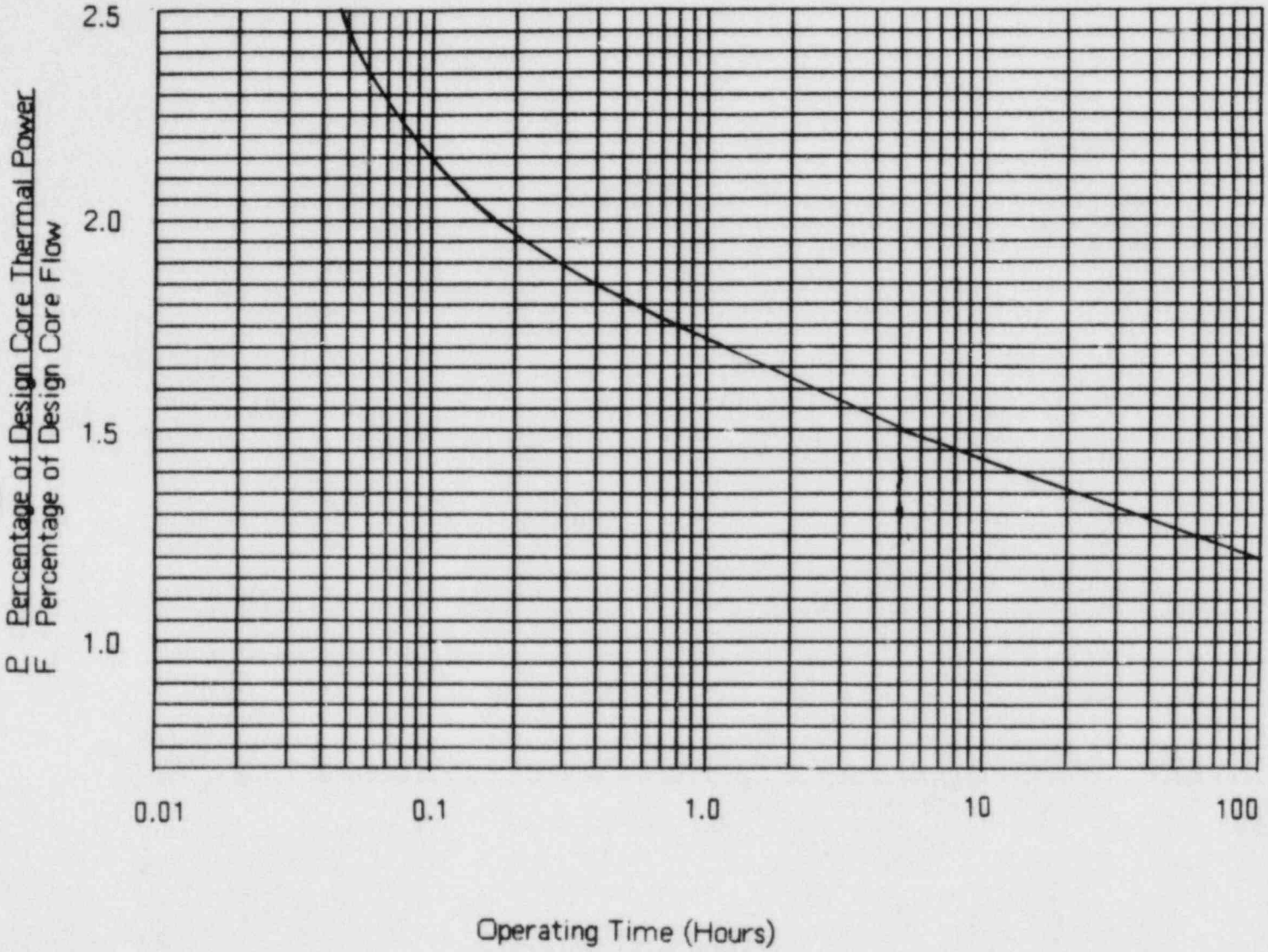
1) As soon as practicable, but no more than 12 hours after any transient exceeding the curve of Figure 2.1.1-2, evaluate the total Integrated Operating Times of each fuel segment within the core at POWER-TO-FLOW RATIOS above the curve of Figure 2.1.1-2. These Integrated Operating Times will be compared with the Integrated Operating Time Limits of Specification 2.1.1.1, 2.1.1.2, 2.1.1.3 and 2.1.1.4 to assure that an Integrated Core Safety Limit has not been exceeded.

2) If an Integrated Operating Time limit of Specification 2.1.1.1, 2.1.1.2, 2.1.1.3, or 2.1.1.4 is exceeded, initiate an orderly SHUTDOWN within 12 hours of exceeding the limit, be in STARTUP within the next 12 hours and be shutdown within the subsequent 12 hours, and comply with the requirements of Specification 6.7.

3) If an Integrated Operating Time Limit and an Individual Transient Operating Time Limit are exceeded simultaneously, the more restrictive Action Statement shall apply.

DRAFT

APR 1 1985

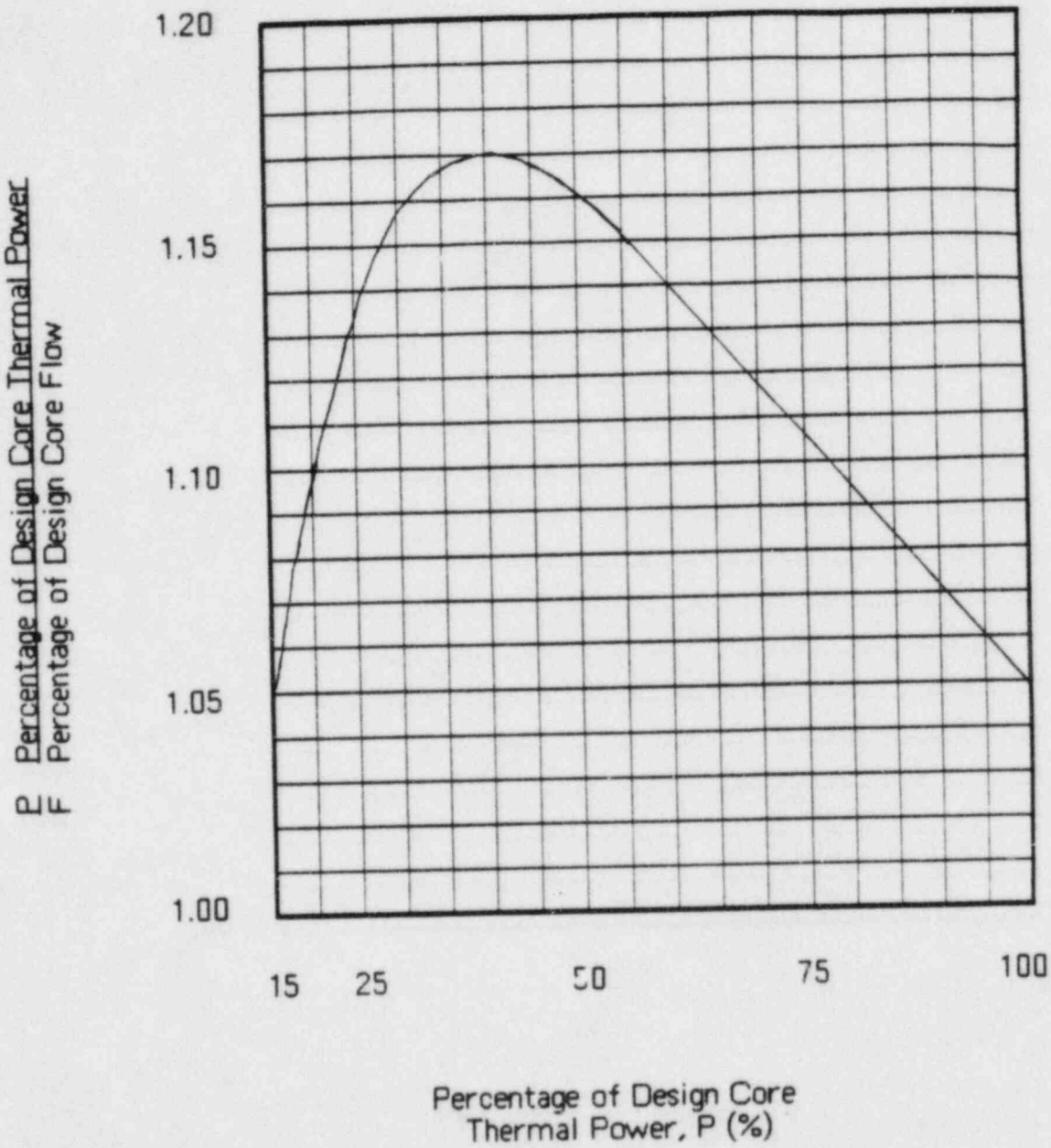


CORE SAFETY LIMIT

Figure 2.1.1-1

DRAFT

APR 1 1985



CORE SAFETY LIMIT

Figure 2.1.1–2

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION SL 2.1.1

To assure integrity of the fuel particles as a fission product barrier, it is necessary to prevent the failure of significant quantities of fuel particle coatings. Failure of fuel particle coatings can result from the migration of the fuel kernels through their coatings. During power operation, there is a temperature gradient across each fuel rod, the higher temperature being at the center of the fuel rod and the lower temperature at the outer edge of the fuel. In an overtemperature condition, fuel kernels can move through their coatings in this temperature gradient, in the direction of the higher temperature.

The Reactor Core SAFETY LIMIT has been constructed to assure that a fuel kernel migrating at the highest rate in the core will penetrate a distance less than the combined thickness of the buffer coating plus the inner isotropic coating on the particle.

The fraction of failed particle coatings in the core at all times is determinable by measurement of gaseous fission product activity in the primary loop.

Calculational Methods and Assumptions for Curves of Figures 2.1.1-1 and 2.1.1-2

In Figures 2.1.1-1 and 2.1.1-2, the quantity P (percentage of RATED THERMAL POWER) is Core THERMAL POWER (MW) divided by 842(MW), and multiplied by 100%. The quantity F (Percentage of Design PRIMARY COOLANT FLOW) is the total coolant flow measured at the circulators (in lb/hr) divided by 3.5E+06 lb/hr, and multiplied by 100%.

The limiting combinations of Core THERMAL POWER and core coolant flow rate are established using a series of short time conservative assumptions. All hot channel factors discussed in Section 3.6 and all power peaking factors discussed in Section 3.5.4 of the FSAR were applied in determining this limiting curve. The range of region radial power peaking factors was assumed to be less than or equal to 1.83 and greater than or equal to 0.4. The maximum intra-region power peaking factor (average power density in a fuel column, Pcol, divided by the average power density in a fuel region, Preg) used was 1.46 plus or minus 0.2 for regions with control rods inserted and 1.34 plus or minus 0.2 for all unrodded regions. A conservative estimate of the most unfavorable axial power distribution was also used. That is, the ratio of power density in the bottom layer of fuel elements of a core region, P lower layer, to the average power density of the region, Preg, is less than or equal to 0.90 plus or minus 0.09 for regions with control rods fully inserted or withdrawn, and 1.23 plus or minus 0.12 for regions with control rods inserted more than two feet.

DRAFT

APR

BASIS FOR SPECIFICATION SL 2.1.1 (Continued)

The measured INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE for the nine regions with their orifice valves most fully closed and all regions with control rods inserted more than two feet, was assumed to be not more than 50 degrees F greater than the CORE AVERAGE OUTLET TEMPERATURE. The measured INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE for the remaining core regions was assumed to be not more than 200 degrees F greater than the CORE AVERAGE OUTLET TEMPERATURE. A measurement uncertainty for the core region outlet temperature of plus or minus 50 degrees F was assumed. A 5% uncertainty in flow measurement and a 5% uncertainty in reactor THERMAL POWER measurement was assumed in establishing the limit. The dependence of the rate of migration of the particle kernel upon temperature and temperature difference across the particle kernel using 95% confidence levels on the experimental data was used.

For the total fuel lifetime in the core, based on calculations incorporating plant parameters and uncertainties appropriate for longer times, migration of the fuel particle kernel through its coating would be less than 20 microns for the fuel with the most damaging temperature history and with the core operated constantly at any of the POWER-TO-FLOW RATIOS and power combinations shown on the curve of Figure 2.1.1-2. Out of a total inner coating thickness of 70 microns, only 50 microns have been used for the determination of fuel particle failure in setting the limit curve in Figure 2.1.1-1.

As can be seen from Figure 2.1.1-1, sufficient time (at least nine minutes) is available for the operator to take corrective action to prevent the core safety limit from being exceeded for POWER-TO-FLOW RATIOS less than or equal to 2.0. To reach a POWER-TO-FLOW RATIO of this magnitude through an increase in core power, significant equipment malfunction, or failure, and/or one or more significant deviations from operating procedures would have to occur.

Bases for POWER-TO-FLOW RATIOS Above 2.5

However, high core POWER-TO-FLOW RATIOS can also be obtained as a result of a reduction or loss of primary coolant circulation. The core negative coefficient of reactivity provides an instrinsic means to reduce the core power and the POWER-TO-FLOW RATIO, and the plant control system will usually initiate scram sequences in such cases. Nevertheless, for brief periods of time prior to or during the scram, high POWER-TO-FLOW RATIOS can exist. Due to the slow thermal response of the core as a result of its high heat capacity, these POWER-TO-FLOW RATIOS can exist for short periods of time without significantly increasing fuel temperatures and fuel kernel migration distances.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION SL 2.1.1 (Continued)

The behavior of the core during numerous transients has been discussed in the FSAR. The slow thermal response of the core is evident from the analysis results shown in Chapter 14 and Appendix D. For example, the Loss of Forced Circulation (LOFC) accident analysis presented in FSAR Appendix D shows that the maximum core temperature rises at a rate of only 6 degrees F/minute for the first two hours following transient initiation. During that time, however, the primary flow rate is zero, while due to fission product decay heat the effective core power is as high as 3%. Thus, the POWER-TO-FLOW RATIO is far above the highest value shown in Figure 2.1.1-1.

Under fast transient conditions, either abnormal rapid power increases or sudden flow decreases, the allowable Integrated Operating Times of Figure 2.1.1-1, which were derived from steady state calculations, are not a meaningful indicator of kernel migration and fuel integrity, i.e., they are overly conservative. Accordingly, a delay period before kernal migration is appropriate. This delay period represents the time required for the fuel to heat up from normal operating temperatures to the steady state temperatures at higher POWER-TO-FLOW RATIOS represented by the Core Safety Limit Curve. Therefore, this delay period can be allowed without compromising the integrity of the fuel. As a result of many transient analyses, the delay period has been conservatively set at 120 seconds if the POWER-TO-FLOW RATIO is greater than 1.17 but less than 2.5, 100 seconds for transients resulting in a POWER-TO-FLOW RATIO above 2.5 but less than or equal to 15, and 60 seconds if the POWER-TO-FLOW RATIO is greater than 15.

The allowable Integrated Operating Time, after the delay time, for all transients which lead to a POWER-TO-FLOW RATIO in excess of 2.5 is set at 2 minutes, which is also the allowable time, after the delay time, for a POWER-TO-FLOW RATIO of 2.5 given by Figure 2.1.1-1.

Bases for POWER-TO-FLOW RATIOS Below 1.17

The limitation of allowable Integrated Operating Time to a value of 100 hours for all operations with a POWER-TO-FLOW RATIO above the curve of Figure 2.1.1-2 and below a value of 1.17 provides a conservative limit since this is the allowable time for a POWER-TO-FLOW RATIO of 1.17 given by Figure 2.1.1-1. Based on the analyses, a 4 (four) hour transient Operating Time in this range of POWER-TO-FLOW RATIOS would be conservative with reference to possible fuel damage. However, from an operating viewpoint, a 30 minute Transient Operating Time Limit has been established as adequate for operator action, which adds sufficient conservatism. If the transient is not reduced below Figure 2.1.1-2 within 30 minutes, an orderly shutdown is appropriate.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION SL 2.1.1 (Continued)

Basis for Applicability

"APPLICABILITY" is limited to power levels above 15% of RATED THERMAL POWER, in that Figure 2.1.1-2 covers only the range of 15% to 100% power. Specification 3.2.4, "Core Inlet Orifice Valves/Minimum Helium Flow", applies to power levels below 15%, where core temperatures are lower.

Basis of 12 Hours for Integral Evaluation in Action c. 1)

Following a deviation from the normal power-to-flow operating range, up to 12 hours are allowed by action statement c. 1) to perform an evaluation, determine if an Integrated Operating Time Limit of Specification 2.1.1 has been exceeded during the deviation, and initiate an orderly shutdown. Twelve hours is considered a sufficient period for determining if the added increment to Integrated Operating Time caused by a transient has caused a conservatively determined Integrated Operating Time limit of Specification 2.1.1 to be exceeded. Shutdown is allowed to be performed in an orderly manner (12 hours to be in STARTUP and an additional 12 hours to be in SHUTDOWN), thus minimizing unnecessary transient effects on other plant components. Any severe transient that causes Specification 2.1.1 to be substantially exceeded would cause a much faster plant shutdown per ACTION statement b, if not by automatic response of the Plant Protective System.

DRAFT

APR 1 1985

2.1 SAFETY LIMITS

REACTOR VESSEL PRESSURE

2.1.2 Neither the PCRV internal pressure nor penetration interspace pressures shall exceed the REFERENCE PRESSURE of 845 psig.

APPLICABILITY: At all times

ACTION:

Whenever the PCRV internal pressure or penetration interspace pressure has exceeded 845 psig, be in SHUTDOWN within 5 minutes with the pressure within its limit within 1 hour and comply with the requirements of Specification 6.7.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION SL 2.1.2

The restriction of this Safety Limit protects the PCRV and its penetrations from overpressurization, and thereby assures the integrity of the PCRV as a fission product barrier.

See FSAR Section 5 for a detailed discussion of the PCRV and penetrations, including design bases. All pressure containing elements for the primary coolant system are designed, with a design pressure equal to the PCRV REFERENCE PRESSURE of 845 Psig. (See FSAR Section 5.8.2). From initial reactor startup, after completion of the Initial Proof Test, pressurization of the PCRV above REFERENCE PRESSURE is positively prevented by means of the safety valve installation, described in Section 6.8 of the FSAR and in Specification 2.2.1. The safety limit of 845 psig is therefore consistent with the design criteria.

Prior to initial operation, the Fort St. Vrain PCRV was subjected to the initial proff test pressure (approximately equal to 1.15 times Reference Pressure) to demonstrate integrity, to verify the structural response of the vessel to an internal pressure greater than REFERENCE PRESSURE, and to demonstrate at an early age that the PCRV, when pressurized to the REFERENCE PRESSURE level, will remain in a net compressive condition at the end of design life.

NOTE: See Specification 3.9.1 for pressure limits applicable during irradiated fuel handling in the reactor vessel.

DRAFT

APR 1 1985

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

TRIP SETPOINTS:

2.2.1 The PLANT PROTECTIVE SYSTEM instrumentation and PCRV Pressurization Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: PLANT PROTECTIVE SYSTEM: As shown for each channel in Table 3.3.1.1

PCRV PRESSURIZATION SETPOINTS: As shown in Specification 3.6.1.

ACTION: With a LIMITING SAFETY SYSTEM SETTING less conservative than the value shown in the Allowable Value column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1, Plant Protective System, or 3.6.1, PCRV Pressurization.

DRAFT

APR 1 1985

Table 2.2.1-1

LIMITING SAFETY SYSTEM SETTINGS

PARAMETER	FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Core Limiting Safety System Settings			
a) Linear Channel-High (Neutron Flux)	Scram	Varies as a Function of Indicated Thermal Power (a)	Varies as a Function of Indicated Thermal Power (a)
b) Reheat Steam Temperature-High	Scram	≤ 1055 degree F	≤ 1061 degree F
c) Primary Coolant Pressure-Programmed Low	Scram	≤ 64.6 psi below normal programmed with Circulator Inlet Temperature. Upper trip setpoint of ≤ 630.6 psia.	≤ 67 psi below normal programmed with Circulator Inlet Temperature per Figure 2.2.1-1, Upper limit to produce trip at ≤ 633 psia.

DRAFT

APR 1 1985

Table 2.2.1-1 (Continued)

LIMITING SAFETY SYSTEM SETTINGS

PARAMETER	FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. Reactor Vessel Pressure Limiting Safety System Settings			
a) Primary Coolant Pressure-Programmed High	Scram and Preselected Loop Shutdown and Steam/ Water Dump	≤ 44 psi above normal programmed with Circulator Inlet Temperature Upper Trip Setpoint of ≤ 744 psia. Lower Trip Setpoint of ≤ 589 psia.	≤ 47 psi above normal programmed with Circulator Inlet Temperature per Figure 2.2.1-1. Upper limit to produce trip at ≤ 747 psia. Lower limit to produce trip at ≤ 592 psia.
b) Primary Coolant Moisture-High	Scram, Loop Shutdown, and Steam/ Water Dump	≤ 60.5 degree F dewpoint temperature	≤ 60.5 degree F dewpoint temperature
c) PCRV Pressure:	Pressure Relief		
Rupture Disc (Low Set Safety Valve)		812 psig plus or minus 8 psig	820 psig

DRAFT

APR 1 1985

Table 2.2.1-1 (Continued)

LIMITING SAFETY SYSTEM SETTINGS

PARAMETER	FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
Low Set Safety Valve		796 psig plus or minus 8 psig	804 psig
Rupture Disc (High Set Safety Valve)		832 psig plus or minus 8 psig	840 psig
High Set Safety Valve		812 psig plus or minus 8 psig	820 psig
d) Helium Circulator Penetration Interspace Pressure:	Pressure Relief		
Rupture Disc (2 Per Penetration)		825 psig plus or minus 17 psig	842 psig
Safety Valve (2 Per Penetration)		805 psig plus or minus 24 psig	829 psig
e) Steam Generator Penetration Interspace Pressure:	Pressure Relief		
Rupture Disc (2 For Each Steam Generator)		825 psig plus or minus 17 psig	842 psig
Safety Valve (2 For Each Steam Generator)		475 psig plus or minus 14 psig	489 psig

DRAFT

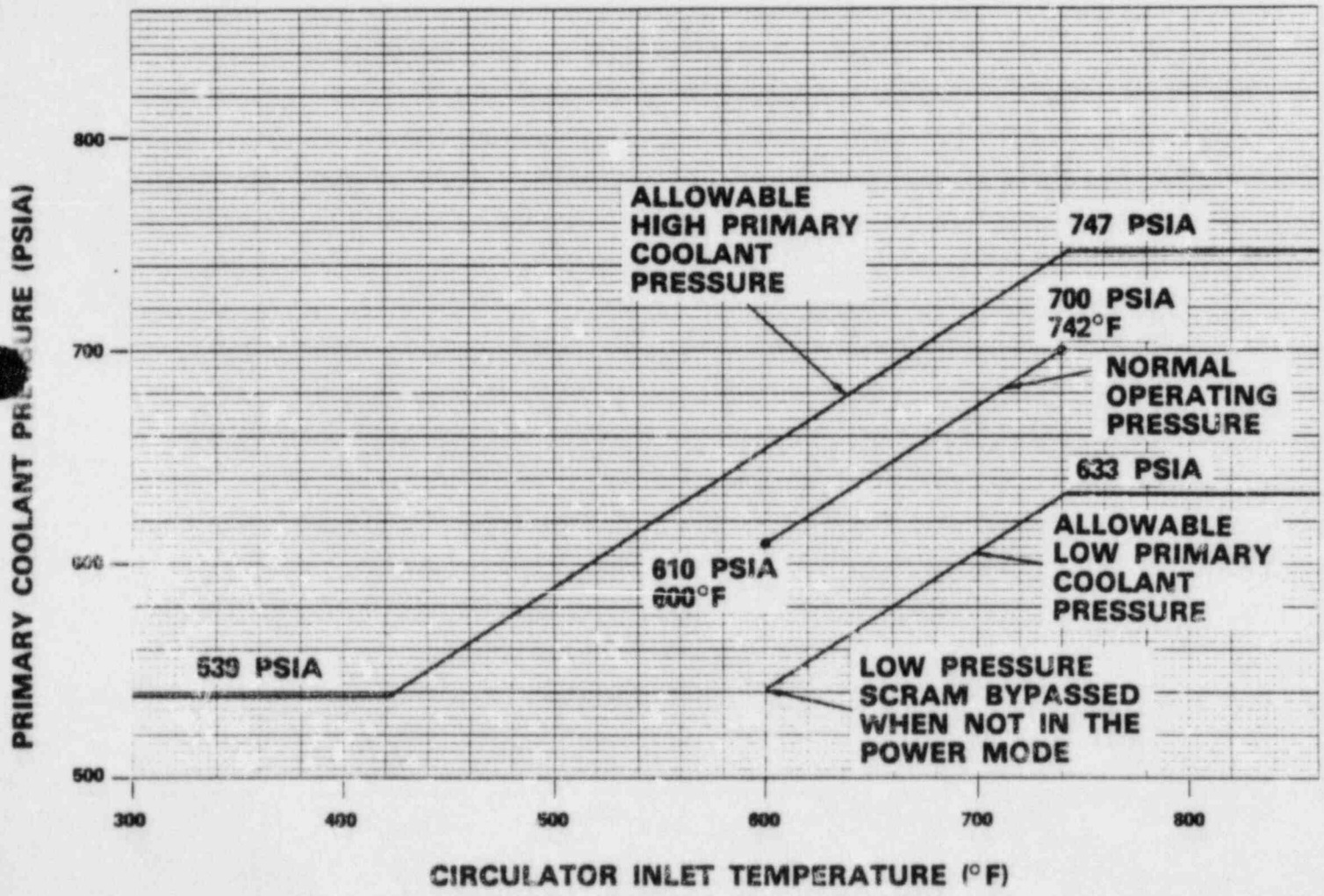
APR 1 1985

Table 2.2.1-1 (CONTINUED)

TABLE NOTATION

(a) Curves specifying the Linear Channel High Neutron Flux Trip Setpoint Limits and Allowable Value as a function of indicated power level, which account for neutron detector decalibration, shall be established for each fuel cycle. The neutron detector decalibration curves shall be approved by the NFSC prior to each fuel cycle. The detector decalibration curves approved by the NFSC shall be forwarded within 30 days of approval to the Regional Administrator, Region IV, Nuclear Regulatory Commission. See Tables 3.3.1-1 and 3.3.1-4 for related limits.

DRAFT
APR 1 1985



PRIMARY COOLANT PRESSURE vs. CIRCULATOR INLET TEMPERATURE
ALLOWABLE OPERATION

Figure 2.2.1–1

BASIS FOR SPECIFICATION SL 2.1.1

TRIP SETPOINTS

Safety Limits established in Specification 2.1 to safeguard the fuel particle integrity and the reactor coolant system barriers. Protective devices included in Specification 2.2 have been provided in the plant design to ensure that automatic corrective action is taken when required to prevent the Safety Limits from being exceeded during normal operation, or during operational transients resulting from possible operator errors, or as a result of equipment malfunction. This specification establishes the Trip Setpoints and Allowable Values for these automatic protective devices.

Linear Channel - High (Neutron Flux)

The neutron flux trip setpoints are established to protect the fuel particle integrity during rapid overpower transients. The power range nuclear channels respond to changes in neutron flux. During normal power operation, the channels are calibrated using a plant heat balance so that the neutron flux that is sensed is read out as percent of RATED THERMAL POWER. For slow maneuvers, those where core thermal power, surface heat flux, and the power transferred to the helium follow the neutron flux, the power range nuclear channels will indicate reactor thermal power. For fast transients, the neutron flux change will lead the change in power transferred from the core to the helium due to the effect of the fuel, moderator and reflector thermal time constants. Therefore, when the neutron flux increases to the scram trip setpoint rapidly, the percent increase in heat flux and power transferred to the helium will be less than the percent increase in neutron flux. Trip setpoints that assure a reactor scram at no greater than 140% of RATED THERMAL POWER are sufficient for the plant because the negative temperature coefficient of reactivity and large heat capacity of the reactor limit the transient increases in fuel and helium temperatures to acceptable values. Control rod shim bank movement can result in decalibration of the ex-core flux detectors. To account for this potential decalibration, the actual trip setpoint is administratively set less than 140% of RATED THERMAL POWER based upon indicated power. These administratively set flux trip setpoints assure the scram will occur at or less than 140% of RATED THERMAL POWER for those postulated reactivity accidents evaluated in FSAR Section 14.2.

BASIS FOR SPECIFICATION SL 2.2.1 (continued)

Reheat Steam Temperature - High

High reheat steam temperature indicates either an increase in thermal power generation without an appropriate increase in helium cooling flow rate or a decrease in steam flow rate. Reheat steam temperature in lieu of reactor core outlet helium temperature is used because of the difficulty in measuring gross helium temperature for protective system purposes. The design of the steam generator is such that changes in hot helium temperature due to a power increase first affect the reheat steam temperature thus allowing the latter to serve as an index of the helium temperature. A reheat steam temperature scram is provided to prevent excessive ratio of power-to-helium flow due to a power increase or steam flow imbalance. (Section 14.2 of the FSAR.)

Primary Coolant Pressure - Programmed Low

The low primary coolant pressure trip setpoint has been established to maintain the fuel particle coating integrity due to loss of primary coolant as the result of a coolant leak.

Primary Coolant Pressure - Programmed High

The major potential source of primary coolant pressure increase above the normal operating range is due to water and/or steam inleakage by means of a defective evaporator-economizer-superheater subheader or tube. For a double ended offset tube rupture, the rate of water and steam inleakage will not exceed 35 lbs/sec initially, resulting in a maximum rate of primary coolant pressure rise of approximately 1 psi per second. The normal plant protection system action upon detection of moisture is reactor scram, loop shutdown, and steam/water dump (FSAR Section 7.1.2.5), occurring after approximately 12 seconds, assuming rated power and flow conditions. In this situation, the peak PCRV pressure at 100% reactor power does not exceed 705 psia. The trip setpoint of $\leq$44 psi above the normal operating pressure between 25% and 100% of rated power is selected: (1) to prevent false scrams due to normal plant transients, and (2) to allow adequate time for the normal protective action (high moisture) to terminate the accident while limiting the resulting peak PCRV pressure in the unlikely event that the normal protective action were inoperative. In this case, reactor pressure would continue to rise to the high pressure trip setpoint. The resulting peak PCRV pressure would be less than the PCRV reference design pressure. The high pressure trip setpoint is programmed as a function of load, using helium circulator inlet temperature as the measured variable indicative of load, as shown in Figure 2.2.1-1. The PCRV safety valves provide the ultimate protection against primary coolant system pressure exceeding the PCRV reference design pressure of 845 psig.

BASIS FOR SPECIFICATION SL 2.2.1 (continued)

Primary Coolant Moisture - High

The high moisture trip setpoint corresponding to 60.5 degree F dewpoint was established, considering the moisture monitor characteristics and the necessity to minimize water inleakage to the reactor system. A trip would be reached after several hours of full power operation with a minimum water/steam inleakage rate in excess of about 20 lbs/hr. Below that inleakage rate, the trip setpoint would never be reached, but the indicating instruments would show an abnormal condition. For maximum design leakage rates, the system behavior is as discussed in the preceding section on High Primary Coolant Pressure. Backup protective action is provided by the high primary coolant pressure scram, loop shutdown, and dump of a pre-selected loop and remaining loop steam depressurization. (FSAR Sections 7.1.2.3 and 7.1.2.4).

PCRV Pressure

If the pressure in the PCRV were to rise significantly above the normal operating pressure, the low-set rupture disc would rupture within the range of 804 psig (-1%), to 820 psig (+1%). The low set safety valve, set at 796 psig plus or minus 1%, would be wide open and flowing full capacity at or above 820 psig (3% accumulation). If the pressure still continued to rise, the high-set rupture disc would rupture between 824 psig and 840 psig. The high-set safety valve, set at 812 psig plus or minus 1%, would be flowing full capacity above 836 psig (3% accumulation). As the pressure decreased, the high-set safety valve would close at a pressure of approximately 690 psig and the low-set safety valve at approximately 677 psig; the corresponding primary system pressure would be approximately 737 psig when the low-set safety valve closed. (FSAR Section 6.8.3). See also Specification 3.6.1.

Helium Circulator Penetration Interspace Pressure

The penetration interspaces are protected against pressures exceeding PCRV reference pressure. The safety valves are set at 805 psig and rupture discs are set at 825 psig (nominal). A redundant safety valve and rupture disc are provided. The rupture discs would burst in the pressure range of 809 psig (-2%) to 842 psig (+2%). The safety valves would open in the range of 781 psig (-3%) to 829 psig (+3%) and would relieve full capacity at 886 psig (10% accumulation). The safety valves would reseat at about 725 psig. The safety valve and rupture disc relieving pressures were specified so as to comply with the ASME Boiler and Pressure Vessel Code, Section III, Class B, Nuclear Vessels, for over pressure protection. See also Specification 3.6.1.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION SL 2.2.1 (continued)

Steam Generator Penetration Interspace Pressure

The six steam generator penetration interspaces in each loop are provided with common upstream rupture discs and safety valves to protect against pressures exceeding PCRV reference design pressure (845 psig). A redundant safety valve and rupture disc are provided. The rupture discs would burst in the pressure range of 809 psig (-2%) to 842 psig (+2%), with a nominal setting of 825 psig. The safety valves are each set at 475 psig which allows for a pressure drop in the inlet lines of 370 psi when relieving at valve capacity. See also Specification 3.6.1.

DRAFT

APR 1 1985

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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APR 1 1985

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

3.0 LIMITING CONDITIONS FOR OPERATION

3.0.1 The Limiting Conditions for Operation specified in this section define the lowest functional capability or performance levels necessary to assure safe operation of the facility. These Limiting Conditions for Operation provide for operation with sufficient redundancy so that further, but limited, degradation of equipment capability or performance, or the occurrence of a postulated incident will not prevent a safe reactor shutdown.

3.0.2 These Limiting Conditions for Operation do not replace plant operating procedures. Plant operating procedures establish plant operating conditions with at least the capability and performance specified in these Limiting Conditions for Operation.

3.0.3 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODES or other conditions specified therein, except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

DRAFT

APR 1 1985

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.0.4 Noncompliance with a Specification shall exist when the requirments of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operations is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.5 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in at least STARTUP within the next 12 hours and in at least SHUTDOWN within the following 12 hours. This Specification is not applicable in SHUTDOWN or REFUELING.

3.0.6 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

3/4.0 APPLICABILITY

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 The Surveillance Requirements specified in this section define the tests, calibrations, and inspections which are necessary to verify the performance and OPERABILITY of equipment essential to safety during designated OPERATIONAL MODES, or required to prevent or mitigate the consequences of abnormal situations.

4.0.2 Surveillance Requirements shall be applicable only during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

DRAFT

APR 1 1985

4.0.3 SURVEILLANCE FREQUENCIES

Where surveillance frequencies or FREQUENCY NOTATIONS are identified, surveillances shall be performed at least once per SURVEILLANCE INTERVAL as specified below. Any surveillance frequency preceded by "at least" shall be considered a nominal time interval to which the extension permitted by Specification 4.0.4 may be applied. Other intervals may be specified in the Surveillance Requirements.

FREQUENCY NOTATION	SURVEILLANCE INTERVAL
S	12 hours*
D	24 hours*
W	7 days*
M	31 days
Q	92 days
SA	184 days
A	366 days
R	Refueling cycle
P	Prior to each reactor startup, if not performed within previous 7 days

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* Two 12 hour SURVEILLANCE INTERVALS shall require performance of the specified surveillances once during any of the AM hours and once during any of the PM hours.

** A 24 hour SURVEILLANCE INTERVAL shall require performance of the specified surveillances once per calendar day.

*** A 7 day SURVEILLANCE INTERVAL shall require performance of the specified surveillances once per calendar week.

DRAFT

APR 1 1985

4.0.4 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. The combined time interval for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.5 Failure to perform a Surveillance Requirement within the specified SURVEILLANCE INTERVAL shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these Surveillance Requirements are stated in the individual specifications. Surveillance Requirements do nat have to be performed on inoperable equipment.

4.0.6 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 have been performed with the stated surveillance interval except surveillances that require entry into the applicable OPERATIONAL MODE shall be performed within 72 hours after entry into that MODE.

4.0.7 The inservice inspection (ISI) surveillance requirements identified in Table 4.0.7-1 shall be per ISI Criterion C and F and shall be implemented prior to STARTUP following the fourth REFUELING CYCLE.

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4.0.8 The SURVEILLANCE INTERVALS for instrumentation and controls may be modified to reflect actual plant experience as specified below:

a. The intervals specified reflect the minimum elapsed time that satisfies these Surveillance Requirements.

b. If the "as found" data on each component or instrument is within the calibration tolerance for 3 consecutive intervals, then the SURVEILLANCE INTERVAL may be extended up to 50% with PORC approval. Applicable to initial SURVEILLANCE INTERVALS less than or equal to A (366 days) but not including the 12 hour, 24 hour or 7 day intervals.

c. If after extending the interval, the "as found" data is out of tolerance for the calibration performed, then the SURVEILLANCE INTERVAL must be shortened to the previous interval, but in no event shorter than the interval specified in the Surveillance Requirements.

DRAFT

APR 1 1985

TABLE 4.0.7-1

ISI SURVEILLANCE REQUIREMENT INSTRUMENTATION

SURVEILLANCE REQUIREMENTS	DESCRIPTION
4.6.1.1.c	PCRV Overpressure Protection
4.6.1.1.d	PCRV S.V. and Rupture Disc Instruments
4.6.4.3.b	PCRV Deformation and Deflections
4.6.4.3.c	PCRV Support Structure Deterioration
4.6.4.5.c.5	PCRV Containment Tank Closure Bolting
4.6.4.5.c.6	PCRV Containment Tank Leak Tightness
4.6.4.5.c.1	PCRV Steam Generator Pentration Welds
4.6.4.5.d.1	PCRV Bottom Access Penetration Welds
4.6.4.5.c.2	Helium Circulator Restraint System
4.6.4.5.d.3	PCRV Safety Valve Penetration
4.6.4.5.d.2	PCRV Bottom Access Penetration Split Split Ring Assembly
4.5.2.1.b.2	Steam Generator Bimetallic Weld Examination
4.5.1.1.b.4	Helium Shutoff Valves
4.5.4.f.2&5	Circulating Water Makeup Pumps

DRAFT
APR 1 1985

REACTIVITY CONTROL SYSTEMS

3/4.1.1 CONTROL RODS

OPERABLE CONTROL RODS/SHUTDOWN MARGIN - OPERATION

LIMITING CONDITION FOR OPERATION

3.1.1.1 A sufficient number of control rod pairs shall be OPERABLE or fully inserted to achieve a SHUTDOWN MARGIN greater than or equal to 0.01 delta k, assuming:

a. A CORE AVERAGE TEMPERATURE of 220 degrees F,

b. All OPERABLE control rod pairs inserted (the regulating control rod pair shall not be considered OPERABLE for the purpose of calculating the SHUTDOWN MARGIN),

c. The decay of Xenon and the buildup of Samarium,

d. The highest worth rod pair is fully withdrawn,

e. The rod worth of any partially inserted inoperable rod pairs at their partially inserted position.

APPLICABILITY: POWER OPERATION, LOW POWER, and STARTUP

ACTION:

a. When a control rod is determined to be inoperable:

1. Within 12 hours achieve full insertion of the rod pair and verify by either: (a) an "in" limit indication and a position indication of plus or minus 10 inches or (b) a watt-meter test, or

2. Determine within 24 hours that the specified SHUTDOWN MARGIN can be met with the rod pair considered inoperable, or

3. Be in at least SHUTDOWN within 36 hours from the time of the determination that the rod pair was inoperable.

DRAFT

APR 1 1985

b. If more than two rod pairs (not including the regulating rod pair) are determined to be inoperable at any one time an engineering review of potential common mode failure mechanisms will be made.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 SHUTDOWN MARGIN VERIFICATION - OPERATING

Verification of SHUTDOWN MARGIN shall be performed when in the POWER OPERATION, LOW POWER or STARTUP as follows:

a. Once per week

b. As required by the ACTION statements of Specification 3.1.1.1, Specification 3.1.2.1, or Specification 3.1.2.2.

4.1.1.1.2 CONTROL ROD PAIR OPERABILITY

Control rod pair OPERABILITY shall be demonstrated as follows:

a. Prior to withdrawal of control rod pairs to achieve criticality (if not performed in the previous week) by a partial scram test of at least 10 inches on all OPERABLE rod pairs. The extrapolated scram time shall be less than or equal to 152 seconds.

b. Weekly when in the POWER OPERATION or LOW POWER MODES by a partial scram test of at least 10 inches on all partially inserted and fully withdrawn control rods except the regulating rod pair. The extrapolated scram time shall be less than or equal to 152 seconds.

c. Daily when in the POWER OPERATION or LOW POWER MODES by a verification that all control rod drive motor temperatures are less than or equal to 272 degrees F. If the motor temperature instrumentation is not available, perform an engineering evaluation to determine that the motor temperatures are less than 272 degrees F.

d. During each refueling outage and during each shutdown with a scheduled duration of 10 days or longer if not performed during the previous month by a full stroke scram test on all control rod pairs. The scram time shall be less than or equal to 152 seconds.

DRAFT

APR 1 1985

e. During each REFUELING CYCLE perform preventive maintenance on control rod drives. The sequencing of this preventive maintence will be such that none of the drives installed in the reactor will have gone more than six REFUELING CYCLES without receiving preventive maintenance.

DRAFT

APR 1 1985

REACTIVITY CONTROL SYSTEMS

3/4.1.1 CONTROL RODS

OPERABLE CONTROL ROD/SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 A sufficient number of control rod pairs shall be fully inserted to maintain a SHUTDOWN MARGIN greater than or equal to 0.01 delta k, assuming:

a. A CORE AVERAGE TEMPERATURE of 80 degrees F.

b. The full decay of Xenon and the buildup of Samarium.

c. The full decay of Pa-233.

d. The highest worth rod pair that is capable of being withdrawn by action of its drive motor is fully withdrawn.

APPLICABILITY: SHUTDOWN and REFUELING

ACTION: With SHUTDOWN MARGIN less than required

a. Suspend all control rod or fuel manipulations involving positive reactivity changes, and

b. Within 8 hours of determining that the specified SHUTDOWN MARGIN is not met, either:

1. Fully insert (as verified by either: (a) an "in" limit indication and a position indication of 0 plus or minus 10 inches or (b) a watt-meter test) sufficient control rods to achieve the specified SHUTDOWN MARGIN, or

2. Actuate sufficient reserve shutdown material to achieve the specified SHUTDOWN MARGIN.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.1.1.2 Verification of SHUTDOWN MARGIN when in the SHUTDOWN or REFUELING shall be performed as follows:

a. As required by the ACTION statement of Specification 3.1.2.1, and

b. Prior to control rod withdrawal if all control rod pairs are not fully inserted prior to the withdrawal action, or

c. Prior to the replacement of fuel in a refueling region, or

d. Prior to control rod withdrawal to achieve criticality to confirm that upon reaching criticality the requirement of Specification 3.1.1.1 can be met (if not performed in the previous week).

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.1.1/SR 4.1.1.

A. SHUTDOWN MARGIN - OPERATING

The purpose of this Limiting Condition for Operation is to assure that during operation a sufficient amount of negative reactivity in control rod pairs is capable of being inserted by the automatic and manual scram functions to shutdown the reactor with the highest worth rod pair fully withdrawn. A SHUTDOWN MARGIN of at least 0.01 delta k has been specified at a CORE AVERAGE TEMPERATURE of 220 degrees F with decay of Xe-135 and buildup of Sm-149. The CORE AVERAGE TEMPERATURE will normally be significantly above 220 degrees F for several days following a scram from power yielding a SHUTDOWN MARGIN greater than 0.01 delta k. In addition the decay of Xe-135 and buildup of Sm-149 occur over a several day period, again causing the SHUTDOWN MARGIN immediately after scram to be larger than the 0.01 delta k specified. Therefore, a twelve hour ACTION combined with a 24 hour delay in verification of the SHUTDOWN MARGIN is sufficient for the purpose of this specification.

For the purpose of this specification, OPERABLE control rod pairs are those withdrawn or partially withdrawn control rod pairs for which periodic surveillance demonstrates both scram capability and valid position indication, thus assuring that the reactivity value of the control rod pairs inserted during the scram is known. Control rod pairs without demonstrated scram capability can be withdrawn and used to control core reactivity via action of the control rod drive motors since control rod pairs which fail to scram by gravity can be positioned by the drive motors. The reactivitiy value of partially inserted control rod pairs without demonstrated scram capability can be included at their partially inserted position in the calculation of the core reactivity. However, no credit can be taken for any additional insertion with regard to available SHUTDOWN MARGIN.

DRAFT

APR 1 1985

Control rod pairs that are verified to be fully inserted are performing their design function and can be included in calculating the SHUTDOWN MARGIN, irrespective of their OPERABILITY. The regulating control rod pair is not considered OPERABLE for the purpose of this specification since its scram capability cannot be verified by surveillance during operation at power.

The allowable number of inoperable, not fully inserted, control rod pairs will depend upon the total SHUTDOWN MARGIN which varies during the REFUELING CYCLE. Measured critical control rod positions along with calculated values of: Xe-135 and Sm-149 worths, control rod pair worths and temperature coefficient are used to calculate the number of inoperable, not fully inserted, control rod pairs which meet this specification.

If more than two control rod pairs are determined to be inoperable at any one time, a review of the causes will be made to determine if a potential exists for a common mode failure which could impact on the OPERABILITY of other control rod pairs.

B. Verification of SHUTDOWN MARGIN-OPERATING

Verification of the SHUTDOWN MARGIN requirements of LCO 3.1.1.1 at least weekly assures that changes in the core reactivity as a result of burnup have not occured which would make the previous verification invalid. The core reactivity changes as a result of burnup occur slowly and weekly surveillance during operation is sufficient. In addition, the ACTION statements of LCO 3.1.1.1, 3.1.2.1, and 3.1.2.2 require more frequent verification if a control rod pair is determined inoperable.

C. Verification of Control Rod Pair OPERABILITY

1. Control Rod Pair OPERABILITY-Partial Scram

The partial scram tests on all rods prior to criticality or weekly on partially inserted and fully withdrawn control rod pairs during POWER and LOW POWER demonstrate that the control rod pairs are capable of being inserted via an automatic or manual scram actuation. Degradation of scram capability occurs over a long term and tests prior to criticality or weekly tests during POWER or LOW POWER are sufficient to identify degradation which could inhibit scram capability.

DRAFT

APR 1 1985

The 152 second scram time is the scram time used in the safety analyses of the FSAR. A nominal scram time of 160 seconds is also identified in the FSAR. Use of the 152 second scram time to determine OPERABILITY is therefore conservative.

2. Control Rod Pair OPERABILITY-Temperature

High control rod drive temperatures which could compromise the operability of control rod drives can only occur when operating in the POWER OPERATION or LOW POWER. Changes in the temperatures of these control rod drives generally occur slowly while at power and a weekly surveillance is sufficient to identify control rod drives which are operating with excessively high temperatures.

Functional testing of the temperature sensors will be performed during their initial installation and during the preventive maintenance of the drives. The 272 degrees F rod drive motor temperature used to determine OPERABILITY is the minimum temperature at which damage may occur to the drive motor or gear train as described in Section 3.8.1.1.2 of the FSAR.

3. Control Rod Pair OPERABILITY-Full Scram

The full stroke scram tests during each refueling outage will supplement the partial stroke weekly scram testing during POWER OPERATION and LOW POWER to assure scram capability of OPERABLE rods.

4. Control Rod Drive Preventive Maintenance

The preventive maintenance on control rod drives will assure that control rod drives are periodically reworked to reduce the impact of long term degradation.

DRAFT

APR 1 1985

D. SHUTDOWN MARGIN - Shutdown

The purpose of this specification is to assure that during REFUELING and SHUTDOWN a sufficient number of control rod pairs are fully inserted to keep the reactor in a shutdown condition. A SHUTDOWN MARGIN of at least 0.01 delta k has been specified at a CORE AVERAGE TEMPERATURE of 80 degrees F with decay of Xe-135, buildup of Sm-149 and decay of Pa-233. The CORE AVERAGE TEMPERATURE will normally be significantly above 80 degrees F for many months after shutdown and the decay of Pa-233 occurs over a few months. Therefore the SHUTDOWN MARGIN immediately after achieving shutdown will normally be larger than the 0.01 delta k specified, and the 8 hour ACTION combined with a 24 hour delay in verification of the SHUTDOWN MARGIN is sufficient for the purpose of this specification.

Experience has shown that a control rod drive which cannot be verifiably demonstrated to be capable of being inserted by scram, can still be inserted under action of the control rod drive motor. Therefore, this specification need only require that the rod pair be actually inserted to achieve the specified SHUTDOWN MARGIN.

The specified SHUTDOWN MARGIN also assumes the full withdrawal of the highest worth rod pair that is capable of being withdrawn by action of its drive motor. The specified SHUTDOWN MARGIN will therefore be maintained considering the accidental withdrawal of the highest worth rod pair and its failure to reinsert by action of the automatic or manual scram functions. Disabling of control rod drives by racking out the drive power at the motor control center results in the inability to withdraw the rod pair by action of the drive motor. Accidental withdrawal of any drive disabled in this manner does not need to be assumed in the evaluation of the SHUTDOWN MARGIN.

E. Verification of SHUTDOWN MARGIN-SHUTDOWN or REFUELING

The ACTION statement of Specification 3.1.2.1 requires completion of the verfication of the SHUTDOWN MARGIN within 24 hours of determining that any fully inserted control rod pair cannot be verfied to be fully inserted. Within the first 24 hours after SHUTDOWN, the SHUTDOWN MARGIN is significantly larger then specified due to higher core temperatures and the presence of Xe-135 and Pa-233. A 24 hour delay will not compromise the validity of Specification 3.1.1.2. Verification of this LCO prior to any control rod withdrawal if all control rods are not fully inserted or prior to removal and insertion of fuel in a refueling region assures that the requirements of Specification 3.1.1.2 will be met during these actions.

DRAFT
APR 1 1985

REACTIVITY CONTROL SYSTEMS

3/4.1.2 CONTROL ROD POSITION INSTRUMENTATION

FULLY INSERTED AND FULLY WITHDRAWN ROD PAIR

LIMITING CONDITION FOR OPERATION

3.1.2.1 For fully inserted or fully withdrawn control rod pairs the position indication system shall be OPERABLE and capable of determining the control rod positions within plus or minus 10 inches by both a position indication and a limit indication.

APPLICABILITY: AT ALL TIMES

ACTION:

a. Any inoperable position indication shall be repaired prior to the return to operation following the next refueling outage.

b. For fully inserted control rod pairs, if full insertion cannot be verified by an "in" limit indication and at least one position indication of 0 plus or minus 10 inches within 24 hours either:

1. Verify full insertion by a watt-meter test, or

2. Identify the rod pair as being inoperable and fully withdrawn in the determination of the SHUTDOWN MARGIN for Specification 3.1.1.1 or 3.1.1.2, and

a. The rod pair shall be identified as being fully inserted for the purpose of Specification 3.1.4 with only one full in position or limit indication, and

b. The discrepancy portion of the Region Peaking Factor Surveillance of Specification 4.2.3 shall be performed within 48 hours and weekly thereafter during operation in this condition while in the LOW POWER and POWER, and

c. Complete the verification of SHUTDOWN MARGIN of Specification 4.1.1.1.1. or Specification 4.1.1.2 depending upon the operational mode.

DRAFT

APR 1 1985

c. For fully withdrawn control rod pairs, if full withdrawal cannot be verified by an "out" limit indication and at least one position indication of 190 plus or minus 10 inches:

1. The rod pair shall be identified as being inoperable for the purpose of determining the SHUTDOWN MARGIN for Specification 3.1.1.1, and

2. The rod pair shall be identified as being partially inserted for the purpose of Specification 3.1.2.2 and 3.1.4, and

3. The discrepancy portion of the Region Peaking Factor Surveillance of Specification 4.2.3 shall be performed within 48 hours and weekly thereafter during operation in this condition while in the LOW POWER and POWER OPERATION, and

4. Complete the verification of SHUTDOWN MARGIN of Specification 4.1.1.1.1 within 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.1 Control rod position instrumentation OPERABILITY shall be demonstrated as follows:

a. Perform a CHANNEL CHECK on the control rod position instrumentation as follows:

1. Prior to withdrawal from the fully inserted position.

2. Upon full withdrawal.

3. At least once per week during all MODES on all control rod pairs except for fully inserted rod pairs which have been disabled by disconnecting motive drive power.

4. After a MODE change to SHUTDOWN.

b. During each REFUELING CYCLE, verify the OPERABILITY of the rod pair redundant "in" and "out" limit switches.

DRAFT
APR 1 1985

REACTIVITY CONTROL SYSTEMS

3/4.1.2 CONTROL ROD POSITION INSTRUMENTATION

PARTIALLY INSERTED ROD PAIR

LIMITING CONDITION FOR OPERATION

3.1.2.2 For partially inserted control rod pairs other than the regulating rod pair, the position indication system shall be OPERABLE and capable of determining control rod position within plus or minus 12 inches.

APPLICABILITY: POWER, OPERATION, LOW POWER and STARTUP

ACTION:

a. When the rod position indications from two separate potentiometers disagree by more than plus or minus 12 inches:

   1. The more fully inserted indication will be used in determining the SHUTDOWN MARGIN for Specification 3.1.1.1, and

   2. The more fully inserted position will be used in Specification 3.1.4, and

   3. The reactivity discrepancy portion of the Reactivity Status Surveillance of Specification 4.1.7 will be performed daily, and

   4. The discrepancy portion of the Region Peaking Factor Surveillance of Specification 4.2.3 will be performed within 48 hours and weekly thereafter while operating in this condition.

b. When only one position indication is available from the two separate potentiometers, within 24 hours and monthly thereafter adjust the reactor power to achieve a fully inserted or fully withdrawn condition and verify the accuracy of the position indication within plus or minus 12 inches of the "in" or "out" limit indication. If this verification can be achieved, the rod pair can be repositioned in the partially inserted position and the ACTION of a.3 and a.4 above will be performed. If this verification cannot be achieved the ACTION of c below will be performed.

DRAFT
APR 1 1985

c. When no position indication is available, within 24 hours:

1. Reduce reactor power as required to achieve full insertion of the rod pair without position indication, and

2. Verify full insertion of the rod pair by watt-meter testing, or perform the ACTION of b.2 under Specification 3.1.2.1.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Control rod position instrumentation OPERABILITY shall be demonstrated by performance of the tests and inspections required by Specification 4.1.2.1.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.1.2. / SR 4.1.2.

A. Fully Inserted Rod Pair

Redundant indication of full insertion of a rod pair is achieved by one full "in" limit indication and one position indication of 0 plus or minus 10 inches. Position indication will normally be provided by the installed digital or analog position indicators or limit lights. Other means can be used to read the position from the position potentiometers or limit switches, if the accuracy of installed indicators or limit lights are in question. An error of plus or minus 10 inches in the full "in" position will not significantly impact on the reactivity value of the rod pair due to the small differential reactivity worth near the fully inserted postion. If these redundant indications are not available the watt-meter test provides a definitive verification of full insertion of the rod pair. Identifying a rod pair which does not have verifiable full insertion as not being fully inserted or OPERABLE for the determination of the SHUTDOWN MARGIN for Specification 3.1.1.1 and 3.1.1.2 results in a conservative determination of the SHUTDOWN MARGIN.

Performance of the discrepancy portion of the Region Peaking Factor Surveillance of Specification 4.2.3 within 48 hours and more frequently during operation in LOW POWER and POWER OPERATION will assure that any partial insertion which would significantly impact upon the core power distribution will be detected in a timely fashion.

B. Fully Withdrawn Rod Pair

Redundant indication of full withdrawal of a rod pair is achieved by one full "out" limit indication and one position indication of 190 plus or minus 10 inches. Position indication will normally be provided by the installed digital or analog position indicators or limit lights. Other means can be used to read the position from the position potentiometers or limit switches, if the accuracy of installed indicators or limit lights are in question. An error of plus or minus 10 inches in the fully withdrawn position will not significantly impact on the reactivity value of the rod pair due to the small differential reactivity worth near the fully withdrawn position. Identifying the rod pair as being inoperable for determining the SHUTDOWN MARGIN for Specification 3.1.1.1 does not take any credit for the negative scram reactivity value of the rod pair.

Performance of the discrepancy portion of the Region Peaking Factor Surveillance of Specification 4.2.3 within 48 hours and more frequently during operation in the LOW POWER and POWER OPERATION will assure that any partial insertion which would significantly impact upon the core power distribution will be detected in a timely fashion.

Identifying the rod pair as being partially inserted for Specification 3.1.2.2 and 3.1.4 assures that any partial insertion of the rod pair will be evaluated in a conservative manner in regards to the core power distribution.

C. Partially Inserted Rod Pair

1. Deviation of position indication greater than plus or minus 12 inches:

   Errors in the position indication of partially inserted rods of up to plus or minus 12 inches will not significantly impact upon the core SHUTDOWN MARGIN or core power distribution.

   If the analog and digital control rod pair position indications disagree by more than plus or minus 12 inches the use of the more fully inserted indicated position in the evaluation of the SHUTDOWN MARGIN for Specification 3.1.1.1 will assure that the reactivity value capable of being inserted upon scram is evaluated in a conservative manner. Any error in control rod position which is significant in terms of its reactivity value or in terms of the core power distribution will be identified in the performance of the Reactivity Status Surveillance of Specification 4.1.7 or the discrepancy portion of the Region Peaking Factor Surveillance of Specification 4.2.3. Performance of these two surveillances on a more frequent basis when the control rod position indication deviate by more than the specified amount will assure that position indication errors which are of significance will be detected in a timely fashion.

   Using the more fully inserted indicated position in assessing conformance with Specification 3.1.4 assures that the Specification is evaluated in a conservative manner.

DRAFT

APR 1 1985

2. With one position indication:

With only one position indication the validity of this indication is verified within 24 hours by comparison with an "in" or "out" limit indication. This verification is repeated monthly thereafter. By performing these comparisons the accuracy of the single position indicator is verified and the partially inserted control rod pair position indication is verified. Performing the reactivity portion of the Reactivity Status Surveillance and the discrepancy portion of the Region Peaking Factor Surveillance on a more frequent basis yields additional verification that any uncertainty in the position indicator will not significantly impact upon the core reactivity or the core power distribution.

If the accuracy of the single position indicator cannot be verified by comparison with an "in" or "out" limit indication the ACTION will be the same as if no position indication is available.

3. With no position indication:

Reduction of the reactor power level as required to achieve full insertion and verification of full insertion within 24 hours will assure that the control rod pair is performing its design function. If verification of full insertion cannot be achieved, performing the ACTION of B.2 under Specification 3.1.2.1 will assure that the SHUTDOWN MARGIN is evaluated in a timely and conservative manner and that any partial insertion which would significantly impact upon the core power distribution will be detected in a timely fashion.

D. Surveillances

The CHANNEL CHECK of control rod position indication prior to withdrawal and upon full withdrawal will identify the OPERABILITY of the position indicators and the "in" and "out" limit switches prior to placing the rod pair in a partially inserted position and after being fully withdrawn from a partially inserted position using fully redundant position potentiometers and limit switches.

The weekly CHANNEL CHECK on all rods will identify the OPERABILITY of the redundant position potentiometers and one of the limit switches for the fully inserted and fully withdrawn rod pairs. For the partially inserted rod pairs only the position potentiometers can be checked. However the check prior to and after being fully withdrawn from the partially inserted position with both the potentiometers and limit switches uses fully redundant instrumentation.

The verification after a MODE change to SHUTDOWN will assure timely evaluation of the requirements of Specification 3.1.2.1.

The above CHANNEL CHECKS will normally be performed using the installed position indicators and the "in" and "out" limit lights. Other means can be used to read the position from the position potentiometers or to verify the "in" or "out" limit condition if required. If a control rod has been verified to be fully inserted by means of a watt-meter test, a CHANNEL CHECK on these rod pairs is not required and the watt-meter test need not be repeated if the rod pair has been continously disabled from being withdrawn since the last watt-meter test by racking out of the drive power at the motor control center.

Verification of the OPERABILITY of the redundant "in" and "out" limit switches during refueling outages assures that undetected failures will not persist for the long term. Verification of the "in" limit switch can only be performed during SHUTDOWN due to the control rod withdrawal sequence requirements of Specification 3.1.4.

DRAFT
APR 1 1985

REACTIVITY CONTROL SYSTEMS

3/4.1.3 CONTROL ROD PENETRATION PURGE FLOW

LIMITING CONDITION FOR OPERATION

3.1.3 Purge flow shall be maintained to each of the eight subheaders of the control rod drive penetrations.

APPLICABILITY: POWER OPERATION*, LOW POWER* and STARTUP*

ACTION:

If subheader purge flow is lost and cannot be restored, be in at least SHUTDOWN within 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3 A CHANNEL CALIBRATION and a CHANNEL FUNCTIONAL TEST of the eight subheader control rod drive purge flow measurement channels shall be performed during each REFUELING CYCLE.

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* With reactor vessel pressure above 100 psia and primary system moisture level greater than 10 ppm total oxidants.

DRAFT

MAR 1 1985

BASIS FOR SPECIFICATION LCO 3.1.3 / SR 4.1.3

The purge flow into the control rod drive assembly floods the control rod drive and limits the upward flow rate of contaminated primary system helium coolant. With primary system oxidant levels greater than 10 ppm, operation without purge flow is limited to 24 hours. If the primary system oxidant level is less than or equal to 10 ppm total oxidants, operation without purge flow is allowable since this is the design environment for the drive.

During the SHUTDOWN or REFUELING the driving force for upward flow of contaminants is significantly reduced and purge flow is not required.

The specified CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST will assure that the instrumentation monitoring subheaders providing purge flow to the control rod drive penetrations is OPERABLE and loss of purge flow is detectable.

DRAFT
APR 1 1985

REACTIVITY CONTROL SYSTEMS

3/4.1.4 CONTROL ROD SEQUENCE AND POSITION REQUIREMENTS

LIMITING CONDITION FOR OPERATION

3.1.4 Control rods shall be withdrawn or inserted in groups in a specified sequence except during scram or rod runback. All rod pairs shall be either fully inserted or fully withdrawn within that sequence except that:

a. Up to two shim groups and the regulating rod pair may be in any position provided the shim groups have an axial separation of at least 10 feet.

b. 6 rod pairs may be inserted up to two feet.

This sequence shall be in compliance with requirements specified in Specification 5.3, REACTOR CORE, and shall be approved by the NFSC.

APPLICABILITY: POWER OPERATION, LOW POWER, and STARTUP

ACTION:

With any control rod pair or group not in compliance, perform the following:

a. Restore the control rods to an acceptable configuration within 4 hours of discovery, or

b. Be in at least STARTUP within the subsequent 12 hours, and SHUTDOWN within the following 12 hours.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.1.4 At the beginning of each REFUELING CYCLE, the reactivity worth of the control rods which are withdrawn from LOW POWER to POWER OPERATION, in the normal withdrawal sequence, shall be measured. The measured group worths shall be compared with the calculated group worths to provide confidence that the calculated criteria upon which the selection of the rod sequence has been based has been satisfied. The measured group worth shall agree with the calculated group worth within plus or minus 20% for all groups except groups 4A and 4D, for which the measured group worth shall be within plus 100%, minus 50% of the calculated group worth.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.1.4 / SR 4.1.4

The specification of a rod pair withdrawal sequence during STARTUP and LOW POWER operation is required to:

a) Assist in evaluating the reactivity worth of rods withdrawn during the approach to critical by indicated changes in the multiplied source neutrons.

b) Ensure that an acceptable power distribution is maintained (peaking factors within design limits) for the condition when many control rods are still inserted.

c) Ensure that the calculated maximum worth rod in STARTUP and LOW POWER operation, if assumed accidentally withdrawn, would result in a transient with consequences no more severe than the rod withdrawal accident analyzed in the FSAR (Section 3.5.3.1 and 14.2.2.7).

The specification of a rod pair withdrawal sequence during POWER OPERATION is required to yield an acceptable power distribution. In addition, the sequence ensures that the combination of maximum single rod pair worth and available core temperature coefficients, in the event of an accidental rod withdrawal, will result in a transient with consequences less severe than that analyzed in the FSAR.

The rod withdrawal accident analysis at rated power, as described in the FSAR, was based on a maximum rod pair worth of 0.012 delta k, using temperature coefficients equivalent to a reactivity defect from refueling (220 degrees F) to operating temperature (1500 degrees F) of 0.028 delta k. For operation in the range from 2 to 100 percent power, the fuel temperature may be lower than the full power operating fuel temperature of 1500 degrees F. This results in a greater number of control rod pairs inserted for the critical configuration, and a larger maximum single rod pair worth. A value larger than .012 delta k for a single rod pair can be safely accommodated if fuel temperatures are lower than 1500 degrees F and/or the temperature defect between refueling temperature (220 degrees F) and operating temperature (1500 degrees F) is greater than .028 delta k. (FSAR Section 14.2.1.1).

DRAFT

APR 1 1985

The specified range of power peaking factors given in Specification 5.3, REACTOR CORE, was used in developing the Reactor Core Safety Limit of Specification 2.1 since the limiting combinations of core thermal power and core coolant flow rate are a function of the region, intra-region and axial power peaking factors. Specifying a control rod withdrawal sequence for each REFUELING CYCLE which has peaking factors within these power peaking factor limits assures that the criteria upon which Specification 2.1 is based are met.

The presence of too many partially inserted rod pairs in the core will tend to push the flux into the bottom half of the core and raise the fuel temperatures. The intra-region and axial power peaking factors used in determining the rod withdrawal sequence for each REFUELING CYCLE will be maintained during normal operation if the rods are inserted and withdrawn in sequence and if partially inserted rods are limited as noted above. (See FSAR Section 3.5.4).

The 6 additional rod pairs which may be inserted up to two feet into the core will permit the operator to move rods to assist in regulating the core region outlet temperatures to those specified in Specification 3.2.2. This has a minimal effect on the axial power distribution, resulting in an increase in the average power density in the lower layer of fuel of less than 5 percent.

The runback function inserts two pre-selected groups of three rod pairs during rapid load reductions (see FSAR Section 7.2.1.2). The partial insertion of these control rod pairs, (FSAR Section 3.5.4.3) in addition to those noted above would increase the average axial power peaking factor in the lower layer of fuel to about 0.85. Negligible fuel particle migration (See 2.1) would occur with this condition in the core for up to four hours.

The ACTION requirement which specifies to be in at least STARTUP within the subsequent 12 hours is prudent, since the core temperatures are significantly reduced at lower power levels. In STARTUP, negligible fuel particle migration would occur as long as the minimum helium flow requirements (Specification 3.2.4) are maintained. It is desirable to reduce plant load and temperatures in a controlled manner.

The measurement of control rod group worths in the normal withdrawal sequence at the beginning of each REFUELING CYCLE will provide an evaluation of calculational methods in determining the control rod group worths in the core configuration for that cycle. The criteria used in selecting the control rod sequence is based on calculated data for the maximum worth for any individual rod pair as well as the calculated peaking factors, (region, intra-region, and axial) in the normal operating control rod configuration. Since the core configuration changes for each REFUELING CYCLE (a new

segment includes approximately one sixth of the total core) this evaluation confirms the ability to predict control rod worths in that specific configuration.

The acceptance criteria for the comparison of measured versus calculated control rod group worth within plus or minus 20% includes an allowance for the calculated uncertainty of plus or minus 10% (FSAR Section 3.5.7.4) and uncertainty in the measurement. A larger acceptance criteria is needed for control rod groups 4A and 4D because of a larger uncertainty in the calculated values. Groups 4A and 4D are five column regions located at the core reflector interface, and the analytical model for control rod worth calculations was developed for seven column regions.

DRAFT

APR 1 1985

3/4.1.5 REACTIVITY CHANGE WITH TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.1.5 The reactivity change due to a fuel temperature increase between 220 degrees F and 1500 degrees F, from refueling temperature to rated power conditions, shall be at least as negative as 0.031 delta k throughout the REFUELING CYCLE.

APPLICABILITY: POWER OPERATION, LOW POWER, AND STARTUP

ACTION: With the temperature defect less negative than that specified, the reactor shall be placed in SHUTDOWN within 12 hours of determination. Reactor operation shall not be resumed until:

a. It can be shown that operation will not occur with temperature coefficients less nagative than those assumed in the FSAR accident analysis, or

b. Nuclear analysis demonstrates that lower temperature coefficients do not adversely impact on the accident analysis in the FSAR and approval is received from the Commission.

SURVEILLANCE REQUIREMENTS

4.1.5 At the beginning of each REFUELING CYCLE the reactivity change as a function of fuel temperature change (temperature coefficient) shall be measured and integrated to obtain the measured reactivity temperature defect. This measured defect shall then be compared to the calculated defect. The measured versus calculated values shall agree within plus or minus 20% to confirm that the calculated reactivity temperature defect between a fuel temperature of 220 degrees F and 1500 degrees F is acceptable in evaluating compliance with the above limit.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.1.5 / SR 4.1.5

The negative temperature coefficient is an inherent safety mechanism that tends to limit power increases during temperature excursions. It is a stabilizing element in flux tilts or oscillations due, for example, to xenon transients.

Fuel temperatures during a power excursion beginning from a high power level are well within design limits regardless of the magnitude of the negative temperature coefficient, provided protective action is initiated by a power level signal. However, if protective action occurs much later, such as from a manual scram or an activiation of the Reserve Shutdown System, peak fuel temperatures will be sensitive to the magnitude of the negative temperature coefficient. Peak fuel temperatures during a power excursion beginning from low (or source) power levels also depend on the negative temperature coefficient, particularly the fuel or Doppler coefficient (FSAR Section 14.2.2).

Requiring a reactivity change at least as negative as 0.031 delta k for a CORE AVERAGE TEMPERATURE increase from 220 degrees F to the 1500 degrees F temperature associated with the nominal RATED THERMAL POWER value, ensures temperature coefficients at least as negative as those used in the FSAR accident analysis. All rod withdrawal transients assumed a reactivity temperature defect of 0.028 delta k which when combined with an uncertainty of plus or minus 10%, yields the specified defect of 0.031 delta k.

The maximum value of reactivity temperature defect occurs at the beginning of the cycle and slowly decreases through the cycle to a minimum value at the end of the cycle. If the calculated value is less than 0.031 delta k at any time in the cycle, operation beyond that time in the cycle requires justification by nuclear analysis that the behavior during the rod withdrawal accident is not jeopardized. However, operation during that portion of the cycle where the reactivity temperature defect is larger than 0.031 delta k is acceptable, but may require additional surveillance.

The major shifts in reactivity change as a function of core temperature change will occur following refueling. The specified frequency of measurement following each refueling will assure that the change of reactivity as a function of changes in core temperature will be measured on a timely basis to evaluate the limit provided in Specification 3.1.5.

DRAFT

APR 1 1985

Since the measurement is made at the beginning of a cycle and the minimum value occurs at the end of a cycle, a direct evaluation cannot be made. However, by comparing the calculated value at the beginning of the cycle with the measured value, and requiring that these agree within plus or minus 20%, an evaluation for compliance can be made using the calculated value at the end of cycle.

DRAFT
APR 1 1985

REACTIVITY CONTROL SYSTEMS

3/4.1.6 RESERVE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.6 At least 6 reserve shutdown (RSD) units of the 7 hopper subsystem and at least 29 units in the 30 hopper subsystem shall be OPERABLE.

APPLICABILITY: POWER OPERATION, LOW POWER and STARTUP

ACTION: With more than one RSD unit inoperable in either subsystem, restore the required number of RSD units to an OPERABLE status within 24 hours or be in SHUTDOWN within the subsequent 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.6 The reserve shutdown system shall be demonstrated OPERABLE:

a. At least once per week by determining that the pressure of the ACM nitrogen bottles, which provide an alternate means of actuating the hopper pressurization valves, is at least 500 psig.

b. At least once per 92 days by:

1. Pressurizing each of the 37 reserve shutdown hoppers above reactor pressure, as indicated by operation of the hopper pressure switch. OPERABLE reserve shutdown hoppers shall be capable of pressurization.

2. Operating the ACM quick disconnect valves.

3. Functionally testing the instrumentation which alarms at low pressure in the reserve shutdown actuating pressure lines. OPERABLE reserve shutdown hoppers shall have an actuating bottle pressure greater than or equal to 1,500 psig.

c. At least once per year by calibrating the test pressurizing gas pressure indicators.

DRAFT

MAR 1 1985

d. At each refueling outage by:

1. Demonstrating that each subsystem is OPERABLE by actuating each group of pressurizing valves from the Control Room. The capability of pressurizing the corresponding hoppers need not be demonstrated during this test. Valve position indication and fail safe operation shall be observed during this test.

2. Calibrating the reserve shutdown hopper pressure switches of those control rod drives removed for preventive maintenance (Specification 4.1.1.2).

3. Visually examining the pipe sections which require disassembly and reassembly within the refueling penetrations, after they have been disassembled for preventive maintenance (Specification 4.1.1.2).

4. Functionally testing two reserve shutdown assemblies off line. One assembly shall contain 20 weight percent boronated material and the other 40 weight percent boronated material. The tests consist of pressurizing the reserve shutdown hopper to the point of rupturing the disc and releasing the poison material.

   The absorber material from the tested hoppers shall be visually examined for evidence of boric acid crystal formation and chemically analyzed for boron carbide and leachable boron content. Failure of a reserve shutdown assembly to perform acceptably during functional testing or evidence of extensive boric acid crystal formation will be reported to the Commission within 30 days per Specification 6.9.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.1.6 / SR 4.1.6

The reserve shutdown system must be capable of achieving shutdown in the event of a situation that prevents the insertion of all control rods.

After extended power operation the reserve shutdown system must cover the temperature defect between 1500 and 220 degrees F, the decay of Xe-135, the buildup of Sm-149, and some decay of Pa-233 to U-233.

The SHUTDOWN requirement of at least 0.01 delta k, with a CORE AVERAGE TEMPERATURE greater than or equal to 220 degrees F, was selected to minimize the reactivity control requirements for the reserve shutdown system and yet maintain an ability to change out a control rod drive (CRD) assembly, if neccesary. Refueling condition requirements are given in Specification 3.9.1, which requires the CORE INLET TEMPERATURE to be less than 165 degrees F. Under normal conditions when the reactor has been operated for several months (which is required for Pa-233 buildup), a CORE AVERAGE TEMPERATURE greater than 220 degrees F is retained for a period of 2-4 weeks even with the CORE AVERAGE INLET TEMPERATURE as low as 100 degrees F. This is adequate time for the replacement of a CRD assembly.

The core reactivity increase due to cooldown and Xe decay occurs fairly rapidly and is worth 0.081 delta k in the initial core and 0.076 delta k in the equilibrium core. During the first 7 days following a shutdown, the core reactivity will rise about 0.002 delta k due to Pa-233 decay minus Sm-149 buildup. If the Pa-233 concentration had reached equilibrium during operation prior to shutdown, the total worth of its decay and Sm buildup would be about 0.030 delta k in the initial core and 0.024 delta k in the equilibrium core.

DRAFT

APR 1 1985

A summary of the calculated reactivity requirements and the calculated worth is given below;

	CALCULATED REACTIVITY REQUIREMENTS for RESERVE SHUTDOWN			CALCULATED WORTH of RESERVE SHUTDOWN		
	Cooldown and Xe Decay delta k	7 days Pa decay Sm Build. and 0.01 delta k	40 days Pa and Sm Build. and 0.01 delta k	Worth 35 Units delta k	Worth 37 Units delta k	Worth 35 Units and Last Rod grp delta k
Initial Core END-CYCLE	.081	.093	.111	.101	.130*	NA
Equilib. Core MID-CYCLE	.076	.088	.102	.088	.120*	.102

* FSAR Section 3.5.3.3 calculated worth in the absence of all control rods.

As seen by the calculated results above, if all 37 units of the reserve shutdown system are inserted, there is sufficient reactivity control to cover core cooldown to 220 degrees F, Xe decay, full Pa decay and Sm buildup. However, with the maximum worth unit inoperable in each subset (subsets of 7 and 30 units), the 35 inserted units were calculated to be worth substantially less, 0.101 delta k for initial core and 0.088 delta k for equilibrium core. This is adequate to allow for 7 days of Pa decay.

The calculated worth for the last control rod group withdrawn in cycles 2, 3 and 4 was 0.0157 delta k, 0.0146 delta k, and 0.0154 delta k. Typically this group will be worth more than 0.014 delta k. At the middle of the cycle, that control rod group would still be inserted. If credit is taken for that control rod group being inserted, there is sufficient reactivity control to allow for about 40 days of Pa decay which is sufficient time to change out several CRD assemblies.

DRAFT

APR 1 1985

Generally, inoperable RSD units are capable of being made OPERABLE within 24 hours. However, in the unlikely event that an inoperable RSD unit cannot be made OPERABLE within this time, it can be seen from the calculated values above, there is adequate time (at least 7 days) following a shutdown using the reserve shutdown system, to allow for corrective action of changing out a CRD assembly.

The middle of the cycle is the most restrictive for the reserve shutdown system for the following reasons;

1. At the beginning of the cycle, more than half of the Pa has decayed because of the refueling, and consequently more control rods would be inserted at RATED THERMAL POWER.

2. Pa equilibrium has been achieved at the middle of the cycle, and the temperature defect for core cooldown is substantially larger than at the end of the cycle.

At power levels less than 5%, sufficient control rods are inserted to augment the calculated worth of the reserve shutdown system to assure the capability of achieving the SHUTDOWN condition.

The reliability of the reserve shutdown system to perform its function will be verified by a control system pressure test and actual off-line rupture tests conducted in the Hot Service Facility or other suitable facility. The control system pressure test demonstrates the ability to pressurize the hoppers and indicates the operability of the control system components. A successful test will increase the hopper pressure above reactor pressure and be indicated by a pressure switch. This differential is well below the minimum 115 psi differential pressure required to burst the disc. The off-line tests consist of actual disc ruptures and poison drops. These will be used to determine the reliability of the differential burst pressure of the disc, and the tendency of the poison material to hang up or deteriorate in the hoppers over extended periods of time.

ACM valve actuation gas may be provided by storage cylinders which can be manually connected to each subsystem valve air header by means of quick-disconnect valves. Availability and OPERABILITY of ACM valve actuation is demonstrated by testing.

DRAFT
APR 1 1985

REACTIVITY CONTROL SYSTEMS

3/4.1.7 REACTIVITY STATUS

LIMITING CONDITION FOR OPERATION

3.1.7 The difference between the observed and expected core excess reactivity, normalized to a calculated base reactivity that has been approved by the NFSC for that cycle, shall not exceed 0.01 delta k.

APPLICABILITY: POWER OPERATION, LOW POWER, and STARTUP

ACTION:

a. With a core reactivity difference between observed and expected greater than 0.01 delta k, the reactor shall be in SHUTDOWN within 12 hours of determination. Operations shall not be resumed until a new base reactivity curve has been approved by the NFSC.

b. All changes to the base reactivity during a cycle shall be approved by the NFSC prior to use and a Special Report describing these changes shall be submitted to the Commission in accordance with Specification 6.9.2 within 30 days.

SURVEILLANCE REQUIREMENTS

4.1.7 At each startup and at least once per 7 days during POWER OPERATION, the reactivity status of the core shall be determined and compared with expected reactivity to ensure that the above limit is satisfied.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.1.7 / SR 4.1.7

An unexpected and/or unexplained change in the observed core excess reactivity could be indicative of the existence of potential safety problems or of operational problems. Any reactivity anomaly greater than 0.01 delta k would be unexpected, and its occurrence would be thoroughly investigated and evaluated. The value of 0.01 delta k is considered to be a safe limit since a SHUTDOWN MARGIN of at least 0.01 delta k with the highest worth rod pair fully withdrawn is always maintained (see Specification 3.1.1).

The normalization to an initial base steady state core condition will eliminate discrepancies due to manufacturing tolerances, analytical modeling approximations and deficiencies in basic data at the beginning of operation. Short term reactivity changes involving the reactivity worth of Sm-149, Xe-135, Pa-233 and fuel temperature can be evaluated explicitly as a function of reactor power/flow history. However, long term reactivity effects involving fuel and lumped burnable poison depletion and fission product poison buildup can only be evaluated via a long term base reactivity curve generated as a function of core burnup. Consequently, to evaluate short term changes in core reactivity, values must be normalized to a base reactivity for the appropriate core burnup.

The specified frequency of the surveillance check of the core reactivity status will assure that the difference between the observed and expected core reactivity will be evaluated regularly.

DRAFT

APR 1 1985

CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

3/4.2.1 CORE IRRADIATION

LIMITING CONDITION FOR OPERATION

3.2.1 The maximum irradiation of the fuel or adjacent reflector elements shall not exceed either of the following conditions:

a. The in-core irradiation lifetime of the fuel elements and reflector elements immediately adjacent to the active core shall be limited to the equivalent of 1800 effective days at RATED THERMAL POWER.

b. The average burnup within a fuel region shall be limited to 110,000 MWD per tonne of initial uranium plus thorium.

APPLICABILITY: POWER OPERATION, LOW POWER and STARTUP

ACTION: If the in-core irradiation lifetime of any fuel element or reflector element adjacent to the active core exceeds the limits of a. above, or the average burnup within a region exceeds the limits noted in b. above, the reactor shall be in SHUTDOWN within 72 hours. Continued operation with those fuel or reflector elements cannot proceed unless an engineering analysis has been submitted to the Commission justifying that fuel particle integrity has been and will be maintained, and that the irradiated lifetime is acceptable.

SURVEILLANCE REQUIREMENTS

4.2.1.1 At least once per 184 days, the in-core irradiation lifetime of all fuel elements and reflector elements adjacent to the active core shall be determined to be less than the above limit.

4.2.1.2 At least once per 184 days, the average burnup within each full region shall be determined to be less than the above limit.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.2.1 / SR 4.2.1

The integrity of the coatings of the fuel particles and graphite dimensional changes is dependent on many variables. Prime variables are the total burnup accumulated by the coated fuel particle and the fast fluence. Limiting the allowable irradiation lifetime and burnup to those specified will ensure that the coated fuel particles and graphite will remain within the demonstrated irradiation test values. Although the burnup and irradiation test results (shown in Appendix A.2 of the FSAR) are generally described in terms of percent FIMA (Fissions per Initial Metal Atom) for both the fissile and fertile particles, these can be readily converted to MWD per tonne of initial uranium and thorium.

Since the in-core irradiation lifetime of the fuel and adjacent reflector elements and the average burnup within a fuel region is not readily determinable by plant operations, it can be determined using the Fort St. Vrain Fuel Accountability System (FSVFAS). This system is updated semi-annually and for each refueling by using the calculated data which is generated for inventory purposes.

DRAFT

APR 1 1985

CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

3/4.2.2 CORE INLET ORIFICE VALVES/REGION OUTLET TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.2.2 The INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE* shall not exceed:

a. With the CORE AVERAGE OUTLET TEMPERATURE greater than or equal to 950 degrees F,

1. The CORE AVERAGE OUTLET TEMPERATURE plus 50 degrees F for the nine regions whose valves are most fully closed and any region with control rods inserted more than two feet.

2. The CORE AVERAGE OUTLET TEMPERATURE plus the limit shown in Figure 3.2.2-1 for the remaining regions.

b. With the CORE AVERAGE OUTLET TEMPERATURE less than 950 degrees F,

The CORE AVERAGE OUTLET TEMPERATURE plus 400 degrees F and the conditions of Specification 3.2.4 must be met for all 37 regions.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION:

a. When an INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE exceeds the above limits by less than 50 degrees F, take corrective action until the out-of-limit condition is corrected. If the out-of-limit condition is not corrected within 24 hours, begin reducing THERMAL POWER and have the condition corrected, or be in STARTUP within the next 12 hours.

*Note: The temperature of any region (20 and 32 thru 37) being monitored by a COMPARSION REGION is as described in the definition of INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE.

DRAFT

APR 1 1985

b. When an INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE exceeds the above limits by greater than or equal to 50 degrees F, but less than 100 degrees F, take corrective action until the out-of-limit condition is corrected. If the out-of-limit condition is not corrected within 2 hours, begin reducing THERMAL POWER and have the condition corrected, or be in STARTUP within the next 12 hours.

c. When an INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE exceeds the above limits by greater than or equal to 100 degrees F, initiate a reduction in THERMAL POWER and take corrective action, until the out-of-limit condition is corrected, or be in STARTUP within 12 hours.

SURVEILLANCE REQUIREMENTS

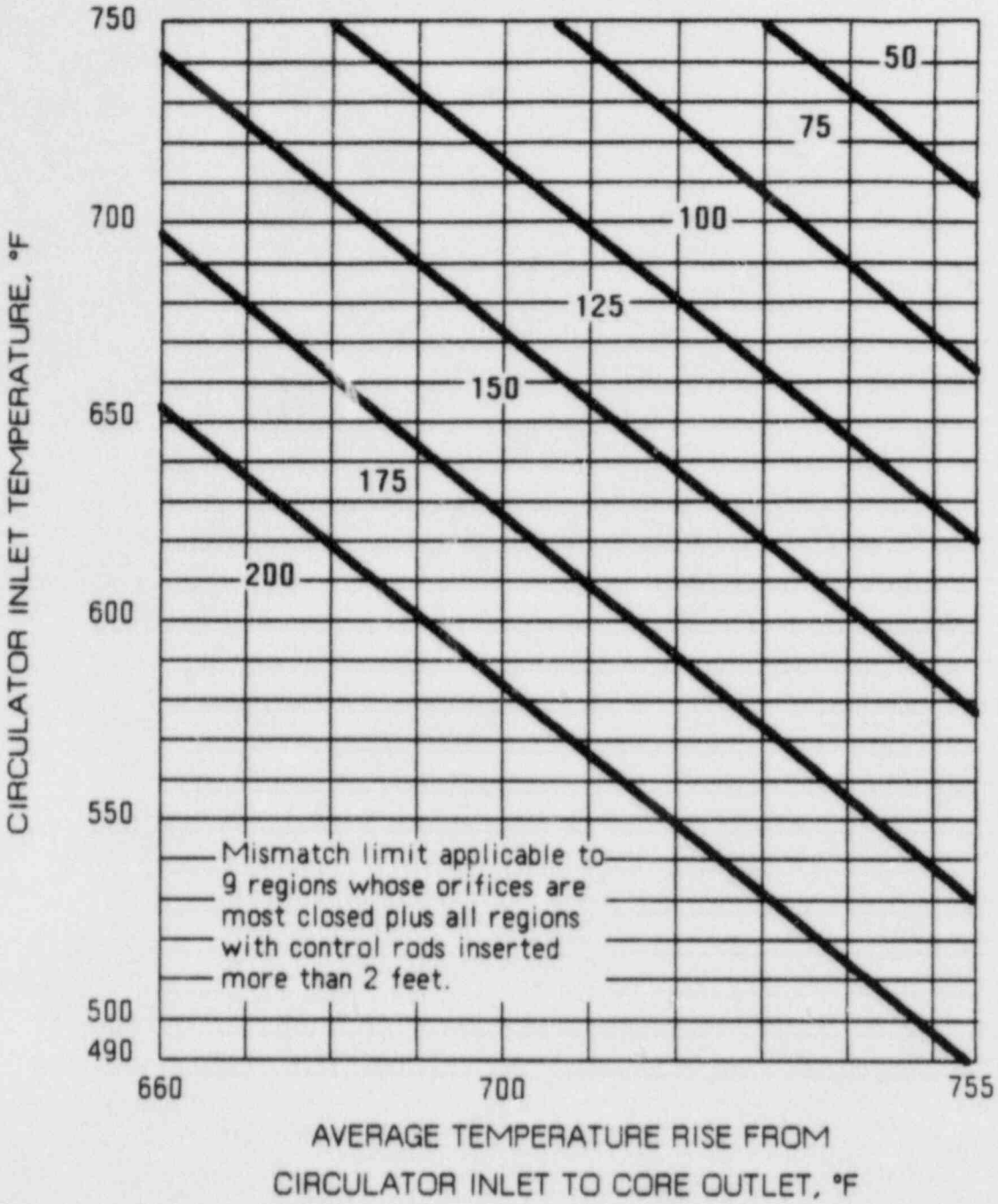
4.2.2.1 All INDIVIDUAL REFUELING REGION OUTLET TEMPERATURES shall be determined to be within the above limits at least once per 8 hours, by obtaining the INDIVIDUAL REFUELING REGION OUTLET TEMPERATURES as determined from either region outlet temperature thermocouples or from the calculated value using the COMPARISON REGION.

4.2.2.2 Each core outlet thermocouple shall be demonstrated OPERABLE:

a. At least once per 24 hours by performing a CHANNEL CHECK.

b. At least once per 18 months by performing a CHANNEL CALIBRATION.

DRAFT
APR 1 1985



Allowable Difference (Mismatch) Between
Region Outlet Temperature and
Core Average Outlet Temperature

Figure 3.2.2–1

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APR 1

BASIS FOR SPECIFICATION LCO 3.2.2 / SR 4.2.2

During the rise-to-power and fluctuation testing above 70% power, the difference between the measured INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE of any region and the CORE AVERAGE OUTLET TEMPERATURE at 100% power was maintained within the limits of Specification 3.2.2. The limits in this specification are more conservative than those used to develop Specification 2.1.1 and those contained in this Specification at the time the testing was conducted. In addition, Specificaiton 3.2.2 directly limits the maximum region outlet temperature to 1,555 degrees F, which is consistent with Table 3.6-1 of the FSAR. By requiring that the limits in Figure 3.2.2-1 be met, maximum fuel temperatures are kept within the FSAR stated values regardless of the power level or the amount of core bypass flow which may exist.

During POWER OPERATION with a CORE AVERAGE OUTLET TEMPERATURE less than 950 degrees F, sufficient over-cooling of the core is provided with a plus 400 degrees F deviation between the maximum INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE and the CORE AVERAGE OUTLET TEMPERATURE to assure that Specification 2.1.1 remains valid and that the integrity of the fuel particles is preserved.

The times allowed for corrective action at temperature exceeding the limits given, represent conditions significantly below the core safety limit determination, Specification 2.1.1.

A COMPARISON REGION is a core refueling region whose power, flow, and coolant outlet temperature characteristics are used to determine the outlet temperature of a region for which the measured INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE is unreliable. Experience has shown that Regions 20 and 32 through 37 have the potential for significant discrepancies between measured and actual INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE. These discrepancies are caused by a transverse flow of relatively cool helium from the core reflector interface along the region outlet thermocouple sleeve. This flow passes over the region outlet thermocouple assemblies of these regions and depresses the indicated outlet temperature.

DRAFT

APR 1 1985

To compensate for these potential transverse flow-induced temperature measurement errors, Regions 20 and 32 through 37 shall have their region outlet temperatures determined by the power and flow characteristics of other regions in the core referred to as COMPARISON REGIONS. The COMPARISON REGION method of operation was first developed for use during rise-to-power and fluctuation testing above 70% power in test procedure RT-500K. Experience obtained during that test indicates that, by use of COMPARISON REGIONS, Regions 20 and 32 through 37 can be operated in a manner consistent with the original reactor design intent and consistent with the criteria upon which Specification 2.1.1 is based.

If the measured region outlet temperature in Regions 20 and 32 through 37 is higher than that based upon the COMPARISON REGION conditions, the measured region outlet temperature is assumed to be correct.

The region outlet temperatures are dependent on power density and coolant flow within each refueling region. The region power density is dependent primarily on the control rod configuration and the coolant flow within each region is controlled by the core inlet orifice valve position. The control rod and orifice valve position configuration is determined by the reactor power level rather than variation in fission product poisons or core burnup which is slowly varying. Therefore, monitoring the region outlet temperatures for changes in reactor power will assure that temperature limits in Specification 3.2.2 can be satisfied.

The long-term thermocouple drift is estimated to be less than or equal to 15 degrees F per year and this drift was included in the measurement uncertainty of plus or minus 50 degrees F used to establish Specification 3.2.2. With this measurement uncertainty, a root mean square difference of greater than or equal to plus or minus 75 degrees F would be an indication of a faulty reading. Daily checks and 18 month calibrations are considered adequate since the expected drift in calibrations is small and has been included in establishing Specification 3.2.2 (see FSAR Section 7.3.3).

DRAFT

APR 1 1985

CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

3/4.2.3 CORE INLET ORIFICE VALVES/COMPARISON REGIONS

LIMITING CONDITION FOR OPERATION

3.2.3 The measured Region Peaking Factor (RPF) shall not be less than 90% of the calculated RPF, for any region used as a COMPARISON REGION.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION: With a measured RPF for a COMPARISON REGION exceeding the above limit:

a. Choose another COMPARISON REGION which is within the limit, or

b. Increase the inferred primary coolant temperature rise (INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE minus the CORE AVERAGE INLET TEMPERATURE) in the region being controlled by the COMPARISON REGION by a percent amount greater than or equal to the amount by which the measured RPF is less than 90% of the calculated RPF.

SURVEILLANCE REQUIREMENTS

4.2.3 The calculated Region Peaking Factors (RPF's) used to determine the INDIVIDUAL REFUELING REGION OUTLET TEMPERATURES for Regions 20 and 32 through 37 and the percent RPF discrepancy for Regions 1 through 19 and 21 through 31 shall be evaluated according to the following schedule for each REFUELING CYCLE:

a. Calculated RPF

1. Prior to initial power operation after each refueling.

2. At 20 plus or minus 5 Effective Full Power Days after each refueling.

3. At 40 plus or minus 5 Effective Full Power Days after each refueling.

DRAFT

APR 1 1985

4. At least once per 31 days thereafter, provided the core has accumulated 10 Effective Full Power Days since the previous evaluation. If the core has not accumulated 10 Effective Full Power Days since the previous evaluation, it may be deferred until the next applicable interval.

b. Percent RPF Discrepancy

1. Above 30% RATED THERMAL POWER but prior to exceeding 40% RATED THERMAL POWER for the first time after each refueling.

2. Within 10 calendar days at reactor power levels above 40% RATED THERMAL POWER after completing any of the calculated RPF evaluations required above. If the total elapsed time at reactor power levels above 40% of RATED THERMAL POWER does not exceed 10 calendar days prior to the subsequent calculated RPF evaluation, the Percent RPF Discrepancy evaluation is not required, but the total elapsed time at reactor power levels above 40% of RATED THERMAL POWER and Percent RPF Discrepancy evaluations shall not exceed 45 calendar days.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.2.3 / SR 4.2.3

Use of COMPARISON REGIONS requires that conditions in the COMPARISON REGION (power, flow, and outlet temperature) be well known. Region peaking factor discrepancies result from combinations of errors or uncertainties in measured region outlet temperature, region flow inferred from orifice valve position, and calculated region power. Based upon an evaluation of data obtained during the rise-to-power testing program, region peaking factor discrepancies up to 10% (positive or negative) are not unexpected or considered to be excessive. Under the COMPARISON REGION method of operation, only excessively negative region peaking factor discrepancies in a COMPARISON REGION could result in prolonged, high fuel temperatures in the region being operated with the COMPARISON REGION.

The requirement that the measured RPF is at least as large as 90% calculated RPF assures that any region being used as a COMPARISON REGION, will not have large negative region peaking factor discrepancies, i.e., are not less than minus 10%.

The % RPF discrepancy is defined as follows:

$$(\%\ \text{RPF discrepancy}) = \frac{\text{RPF measured} - \text{RPF calculated}}{\text{RPF calculated}} * 100\%$$

The calculated region peaking factors for Regions 20 and 32 through 37 and their COMPARISON REGIONS will change during the REFUELING CYCLE as fission product inventories saturate, fissile material and burnable poison are depleted, and control rods are withdrawn from the core. Evaluations based upon operating experience gained prior to completion of rise-to-power testing (i.e., Cycles 1 and 2 and part of Cycle 3) indicate that the ratio of the calculated region peaking factors in Regions 20 and 32 through 37 to the calculated region peaking factors in COMPARISON REGIONS as a function of control rod configuration, changes gradually in a predictable manner during a REFUELING CYCLE. A surveillance check of the calculated region peaking factors at the specified frequency will assure that appropriate region peaking factors continue to be used in determining the region outlet temperature for Regions 20 and 32 through 37.

A surveillance check of the percent region peaking factor discrepancy will provide assurance that the requirements of Specification 3.2.3 are being met for COMPARISON REGIONS. The frequency for surveillance has been established based upon conservative evaluation of potential fuel kernel migration, which could occur if a region with an excessively large, negative region peaking factor discrepancy were used as a COMPARISON REGION.

DRAFT

APR 1 1985

REACTOR CORE AND REACTIVITY CONTROL

3/4.2.4 CORE INLET ORIFICE VALVES/MINIMUM HELIUM FLOW

LIMITING CONDITION FOR OPERATION

3.2.4 a. The reactor helium coolant flow shall be maintained greater than the minimum value shown in Figure 3.2.4-1 for the appropriate power level (including decay heat), when the reactor pressure is greater than 50 psia and the orifice valves are set for equal region coolant flows.

b. The measured helium coolant temperature rise through any core region shall not exceed:

1. The limits given in Figure 3.2.4-2 for the appropriate power level, when the reactor pressure is greater than 50 psia and the orifice valves are not set for equal region coolant flows,

2. 350 degrees F when the reactor pressure is less than 50 psia and the orifice valves are not set for equal region coolant flows.

3. 600 degrees F, when the reactor pressure is less than 50 psia and the orifice valves are set for equal region coolant flows,

APPLICABILITY: LOW POWER*, STARTUP AND SHUTDWON

ACTION:

a. With any of the above limits exceeded, within 15 minutes either:

1. Increase the region helium coolant flow and correct the out-of-limit condition, or

2. Be in at least STARTUP and adjust the inlet orifice valves for equal region coolant flows within 12 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 The total reactor coolant flow or the helium coolant temperature rise through any core region shall be determined to be within the limits of LCO 3.2.4 at least once per 8 hours.

* Applicable only below 15% of RATED THERMAL POWER.

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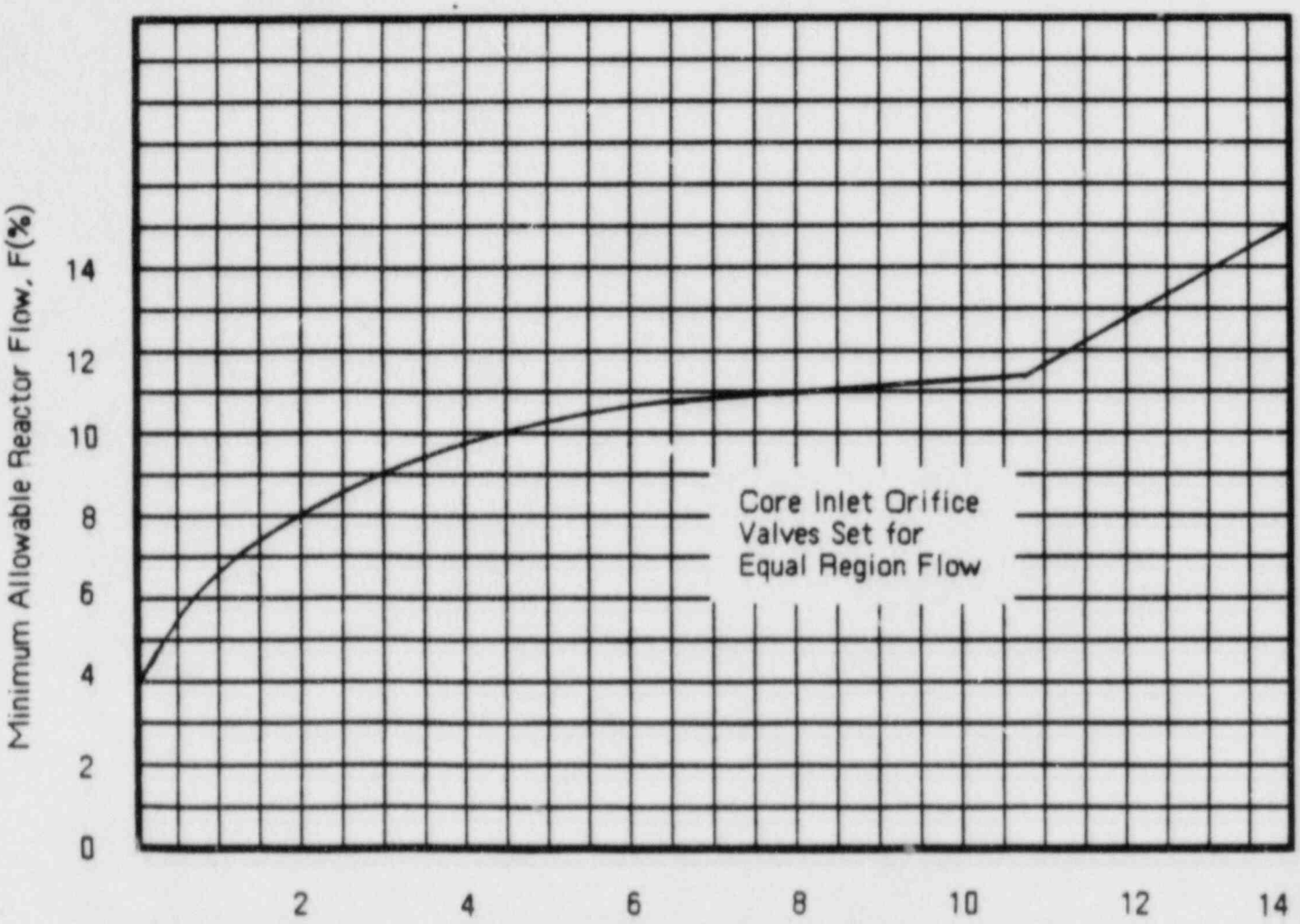
APR 1 1985



Calculated Reactor Thermal Power,

$$P\% = \frac{[\text{Ave. Core Delta T (F)}][\text{Core He Flow (lb/hr)}]}{2.495 \times 10E7}$$

MINIMUM ALLOWABLE REACTOR FLOW

Figure 3.2.4-1

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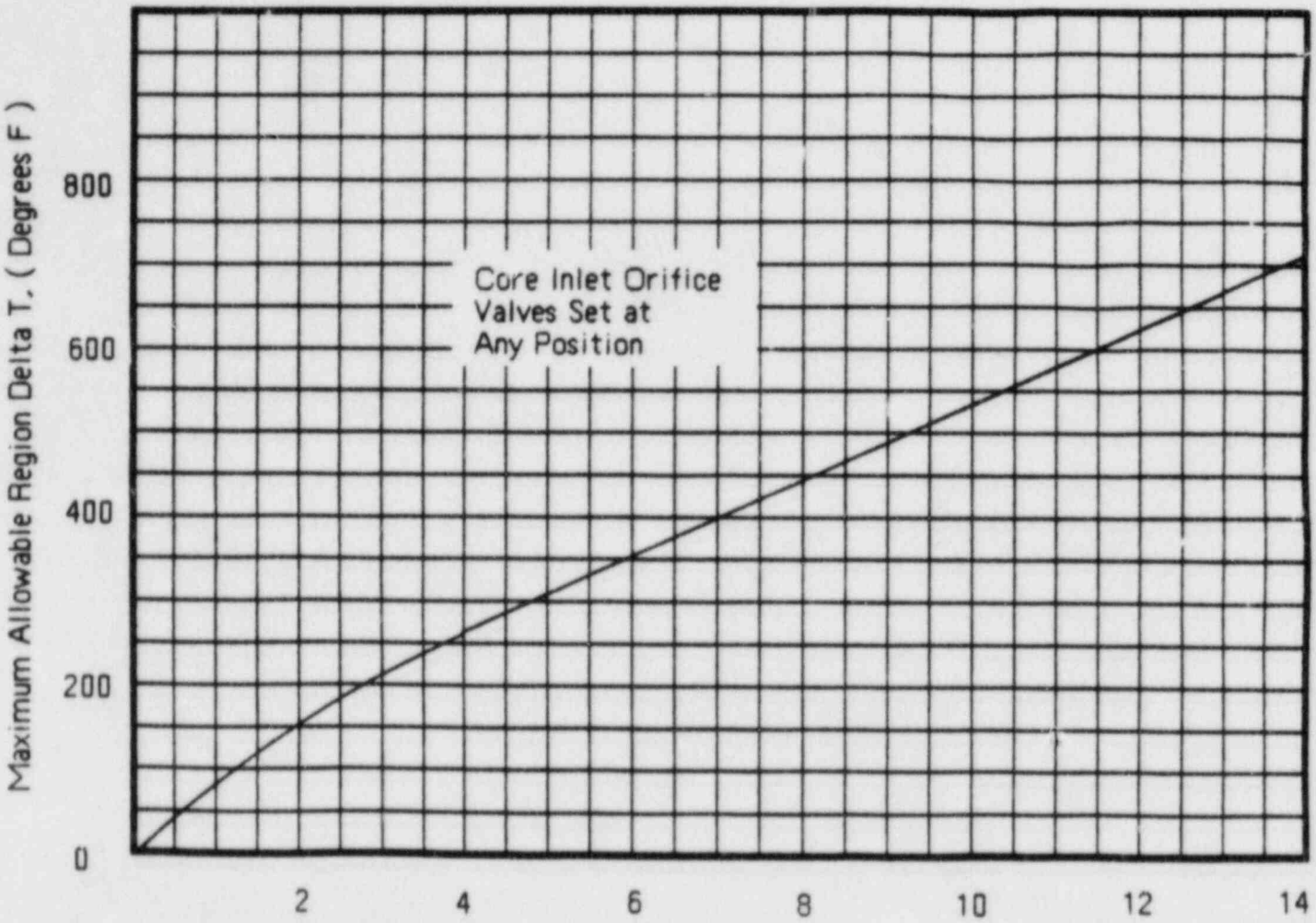
APR 1 1985



Calculated Reactor Thermal Power,

$$P\% = \frac{[\text{ Ave. Core Delta T (F) }][\text{ Core He Flow (lb/hr) }]}{2.495 \times 10E7}$$

MAXIMUM ALLOWABLE REGION DELTA T

Figure 3.2.4–2

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APR 1 1985

BASIS FOR SPECIFICATION LCO 3.2.4 / SR 4.2.4

The minimum reactor helium coolant flow or the maximum core region helium coolant temperature rise as a function of calculated reactor thermal power (including power from decay heat) have been specified to prevent very low helium coolant flow rates through any coolant channel. Very low helium coolant flow rates may result in laminar flow conditions with resultant high friction factors and low heat transfer film coefficients and potential for possible local helium flow stagnation, which could result in excessive fuel temperatures.

The limits have been developed based upon a number of conservative assumptions. It was assumed that the primary system was pressurized to full inventory. At less than full inventory, higher region delta T's are acceptable. The core inlet helium temperature was assumed to be 100 degrees F. At higher core inlet temperatures, higher region delta T's are acceptable. For the condition with helium flow orifice valves adjusted to yield equal helium flow to all fuel elements, a 25% margin of safety was added to the minimum flow requirements. For the condition with orifice valves at any position, the allowable region delta T is based upon a region power density (P-reg/P-core) equal to 0.4. For regions with higher power densities, higher region delta T's are acceptable.

For depressurized operations, limits are also specified to prevent very low helium coolant flow rates through any coolant channel. These limits have been established based upon a 50 psia reactor pressure, and all other conservative assumptions stated above.

This specification addresses minimum flow requirements for all coolant channels. Since low coolant flows exist at lower reactor powers, its applicability is limited to about 15% RATED THERMAL POWER. Above this power level, fuel integrity is ensured by limiting the core region outlet temperatures to those values specified in Specfication 3.2.2. However since THERMAL POWER is continuously generated by decay heat even after the reactor is shutdown, the flow requirements are also applicable in the SHUTDOWN mode.

By monitoring the total reactor coolant flow when the orifices are adjusted for equal region coolant flows, minimum flow through each region at the appropriate power can be assured. When the orifice valves are adjusted to different positions, minimum coolant flows can be assured for each region by monitoring the helium coolant temperature rise in that region.

DRAFT

APR 1 1985

CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

3/4.2.5 REGION CONSTRAINT DEVICES

LIMITING CONDITION FOR OPERATION

3.2.5 The Region Constraint Devices (RCDs) shall be OPERABLE WITH:

a. RCDs in place on top of the core,

b. RCD pins engaged in the fuel columns, and

c. RCD structural integrity maintained.

APPLICABILITY: POWER OPERATION, LOW POWER and STARTUP

ACTION:

a. With one or more RCDs inoperable, shall remain in the SHUTDOWN or REFUELING MODE until the inoperable RCDs are restored to OPERABLE status.

b. Provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.5 The Region Constraint Devices (RCDs) shall be demonstrated OPERABLE for those regions being refueled by:

a. Visually inspecting the upper core plenum to verify that the RCDs within visible range are in place on top of the core.

b. Monitoring the fuel handling machine location coordinates and lifting force as RCDs are being removed, to verify that the RCD pins were engaged in the fuel columns and that they disengage as expected.

c. Visually inspecting at least two selected RCDs from the regions being refueled to verify that there are no abnormal cracks, deformations, loose or missing parts, or other visible defects affecting structural integrity.

DRAFT

APR 1 1985

d. Monitoring the fuel handling machine location coordinates as the RCDs are reinstalled to verify that the RCD pins have engaged in the fuel columns.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.2.5 / SR 4.2.5

Region constraint devices (RCDs), located on top of fuel columns of generally three adjacent fuel regions, restrain region movements in relation to one another by means of centering pins inserted in the handling hole of the upper plenum elements (see FSAR Section 3.3.1.1). The RCDs are used to prevent movement of the fuel columns which mitigates temperature fluctuations in the primary coolant circuit at the individual core region outlets as discussed in Section 3.6.6 of the FSAR.

Visually inspecting the RCDs will ensure that they are providing their design function of restraining the fuel columns.

Monitoring the lifting force to remove the RCDs with the fuel handling machine will provide early indications, should a phenomenon occur over time which might eventually prevent them from moving with the fuel columns or prevent their removal from the reactor.

DRAFT

APR 1 1985

3/4.3 INSTRUMENTATION

3/4.3.1 PLANT PROTECTIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the PLANT PROTECTIVE SYSTEM Instrumentation and 480V Essential Bus Undervoltage Protection channels identified in Part 2 of Tables 3.3.1-1 through 3.3.1-5 shall be operable with Trip Setpoints equal to or greater than (or less than when indicated) the values displayed in Part 1 of Tables 3.3.1-1 through 3.3.1-5.

APPLICABILITY: As shown in Part 2 of Tables 3.3.1-1 through 3.3.1-5.

ACTION:

With a PLANT PROTECTIVE SYSTEM Instrumentation Functional Unit "as found" value less conservative than the value shown in the Allowable Value column of Tables 3.3.1-1 through 3.3.1-5, declare the channel inoperable and apply the applicable ACTION statement requirement.

A channel that has been placed in the tripped condition as a result of an ACTION requirement shall be considered OPERABLE for the purposes of Specification 3.0.6 relative to entry into other OPERATIONAL MODES.

SURVEILLANCE REQUIREMENTS

4.3.1 Each PLANT PROTECTIVE SYSTEM channel and interlock and the automatic trip logic as well as other critical instrumentation and control systems shall be demonstrated OPERABLE by the performance of the PLANT PROTECTIVE SYSTEM Instrumentation Surveillance and Calibration Requirements specified in Tables 4.3.1-1 through 4.3.1-5 and associated notes and as augmented by Specification 4.0.8.

DRAFT
APR 1 1985

Table 3.3.1-1 (Part 1)

INSTRUMENTATION TRIP SETPOINTS FOR PLANT PROTECTIVE SYSTEM, SCRAM

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1a.	Manual Scram (Control Room)	Not Applicable	Not Applicable
1b.	Manual Scram (Outside Control Room)	Not Applicable	Not Applicable
2a.	Start-up Channel High	≤8.3E+04 cps	≤9.3E+04 cps
2b.	Wide Range Channel Rate of Change- High	≤4.5 dpm	≤4.5 dpm
3a.	Linear Channel-High Channels 3,4,5 (Neutron Flux)	Varies as a Function of Indicated Thermal Power (a)	Varies as a Function of Indicated Thermal Power (a)
3b.	Linear Channel-High Channels 6,7,8 (Neutron Flux)	Varies as a Function of Indicated Thermal Power (a)	Varies as a Function of Indicated Thermal Power (a)
4.	Primary Coolant Moisture(b)		
	a) High level Monitor	≤60.5 degree F dewpoint	≤60.5 degree F dewpoint
	b) Loop Monitor	≤20.4 degree F dewpoint	≤20.4 degree F dewpoint
5.	Reheat Steam Temperature -High (b)	≤1055 degree F	≤1061 degree F

DRAFT

APR 1 1985

Table 3.3.1-1 (Part 1) (Continued)

INSTRUMENTATION TRIP SETPOINTS FOR PLANT PROTECTIVE SYSTEM, SCRAM

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
6.	Primary Coolant Pressure -Programmed Low (b)	≤64.6 psi below normal, programmed with Circulator Inlet Temperature. Upper trip setpoint of ≤ 630.6 pisa.	≤67.0 psi below normal, programmed with Circulator Inlet Temperature per Figure 2.2.1-1. Upper limit to produce trip at ≤ 633 psia.
7.	Primary Coolant Pressure -Programmed High	≤44 psi above normal, programmed with Circulator Inlet Temperature Upper Trip Setpoint of ≤ 744 psia. Lower Trip Setpoint of ≤ 589 psia.	≤47 psi above normal, programmed with Circulator Inlet Temperature per Figure 2.2.1-1. Upper limit to produce trip at ≤ 747 psia. Lower limit to trip at ≤ 592 psia.
8.	Hot Reheat Header Pressure -Low	≥44 psig	≥44 psig
9.	Main Steam Pressure-Low	≥1529 psig	≥1529 psig
10.	Plant Electrical System-Loss	≥278V(c)	≥278V(c)
11.	Two Loop Trouble Scram Logic	Not Applicable	Not Applicable
12.	High Reactor Building Temperature (Pipe Cavity)	≤161 degree F	≤165 degree F

DRAFT

APR 1 1985

Table 3.3.1-1 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, SCRAM

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1a.	Manual Scram (Control Room)	2	1	2	P,L,S/U,R (d)	1
1b.	Manual Scram (Outside Control Room)	3	2	2	P,L,S/U,R (d)	2
2a.	Start-up Channel-High	2	1	2	R(d)	3
2b.	Wide Range Channel Rate of Change-High	3	2	2	S/U	2
3a.	Linear Channel-High Channels 3,4,5	3	2	2	P,L,S/U	2
3b.	Linear Channel-High Channel 6,7,8	3	2	2	P,L,S/U	2
4.	Primary Coolant Moisture					
	a) High Level Monitor	2	1(e)	1(f) 1(k)	P,L,S/U	4
	b) Loop Monitor	3	2	2/Loop (f)(g) (k)	P,L,S/U	4
5.	Reheat Steam Temperature -High	3	2	2(h)	P,L,S/U	2

DRAFT
APR 1 1985

Table 3.3.1-1 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, SCRAM

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6.	Primary Coolant Pressure -Programmed Low	3	2	2(i)(j)	P	2
7.	Primary Coolant Pressure -Programmed High	3	2	2(i)(j)	P,L,S/U	2
8.	Hot Reheat Header Pressure -Low	3	2	2	P	2
9.	Main Steam Pressure -Low	3	2	2	P	2
10.	Plant Electrical System-Loss	3	2	2(c)	P,L,S/U	2
11.	Two Loop Trouble Scram Logic	3	2	2	P,L,S/U	2
12.	High Reactor Building Temperature (Pipe Cavity)	3	2	2	P,L,S/U	2

DRAFT

APR 1 1985

Table 3.3.1-1 (Continued)

TABLE NOTATION

(a) Curves specifying the Linear Channel High Neutron Flux Trip Setpoint limit and allowable value as a function of indicated power level, which account for neutron detector decalibration, shall be established for each fuel cycle. The neutron detector decalibration curves shall be approved by the NFSC prior to each fuel cycle. The detector decalibration curves approved by the NFSC shall be forwarded within 30 days of approval to the Regional Administrator, Region IV, Nuclear Regulatory Commission. See Tables 2.2.1-1 and 3.3.1-4 for related limits.

(b) See also Specification 2.2, Table 2.2.1-1.

(c) One channel consists of three undervoltage relays each monitoring a single phase of a 480 VAC essential bus. A channel trip will occur when two of the three undervoltage relays comprising that channel operate after a preset time delay indicating loss of bus voltage. Initiation of a scram requires two of the three undervoltage relays on two of the three 480 VAC essential buses to operate.

(d) The applicability for the refueling mode is anytime the reactor scram is reset and the control rod drive system is capable of rod withdrawal.

(e) The trip of one primary coolant high level moisture monitor accompanied by trips of two of the three loop moisture monitors for either loop will cause full scram.

(f) A primary coolant dew point moisture monitor shall not be considered OPERABLE unless the following conditions are met:

1)

Reactor Power Range	Minimum Sample Flow
Startup to 2%	$\geq$ 1 scc/sec.
Greater than 2% to 5%	$\geq$ 5 scc/sec.
Greater than 5% to 20%	$\geq$15 scc/sec.
Greater than 20% to 35%	$\geq$30 scc/sec.
Greater than 35% to 100%	$\geq$50 scc/sec.

2) Minimum flow of item 1) is alarmed in the control room and the alarm is set in accordance with the power ranges specified.

3) The ambient temperatures indicated by both temporary thermocouples mounted on the flow sensors in penetrations B1 and B3 are less than 185 degree F.

4) Fixed alarms of $\geq$1 scc/sec and $\leq$75 scc/sec are OPERABLE.

DRAFT

APR 1 1985

(g) Permissible Bypass Conditions:

I. Any circulator buffer seal malfunction.

II. Either loop hot reheat header high activity.

(h) Two thermocouples from each loop, total of four, constitute one channel. For each channel, two thermocouples must be OPERABLE in at least one operating loop for that channel to be considered OPERABLE.

(i) One OPERABLE helium circulator inlet thermocouple in an operating loop is required for the channel to be considered OPERABLE.

(j) For programmed trip functions, a channel is inoperable when the channel which provides the programmed Trip Setpoint is inoperable.

(k) In the event that testing of existing PLANT PROTECTIVE SYSTEM (PPS) moisture monitors or installation and testing of new moisture monitors take place, input trip functions to the PPS which cause scram, loop shutdown, circulator trip and steam water dump should be disabled. In addition, the Analytical System moisture monitors shall be utilized to monitor primary coolant moisture during these tests.

During the time that the PPS moisture monitor trips are disabled, an observer in direct communication with the reactor operator shall be positioned in the control room in the location of pertinent instrumentation. The observer shall continuously monitor the primary coolant moisture levels indicated by at least two moisture monitors and the primary coolant pressure indications, and shall alert the reactor operator to any indicated moisture or pressure change.

DRAFT

APR 1 1985

Table 3.3.1-1 (CONTINUED)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the minimum channels operable requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least STARTUP within the next 12 hours and be in SHUTDOWN within the next 12 hours.

ACTION 2 - With the number of OPERABLE channels less than the total number of channels, restore the inoperable channel(s) to OPERABLE status within 12 hours. If the channel(s) is not restored to OPERABLE status within that 12 hours either;

(a) Place the inoperable channel in the tripped condition unless trip of the channel would cause a scram or

(b) If trip of the channel would cause a scram, place the reactor in a MODE where the limit does not apply within the next 12 hours.

ACTION 3 - With the number of OPERABLE channels one less than the minimum channels operable requirement, suspend all operations involving control rod withdrawal or fuel manipulation in the core and terminate incore maintenance per Specification 3.9.1.

ACTION 4 - With the number of OPERABLE channels less than the total number of channels, restore the inoperable channel(s) to OPERABLE status within 12 hours. If the channel(s) is not restored to OPERABLE status within that 12 hours either;

(a) Place the inoperable channel in the tripped condition unless trip of the channel would cause a scram or

(b) In the special case of an inoperable moisture monitor, the moisture monitor input trip functions to the PLANT PROTECTIVE SYSTEM which cause scram, loop shutdown, circulator trip, and steam/water dump may be disabled for up to 72 hours. During the time that the PLANT PROTECTIVE SYSTEM moisture monitor trips are disabled, an observer in direct communication with the reactor operator shall be positioned in the control room in the location of pertinent instrumentation. The observer shall continuously monitor the primary coolant moisture levels indicated by at least two moisture monitors and the primary coolant pressure indications, and shall alert the reactor operator to any indicated moisture or pressure change or

DRAFT

APR 1 1985

TABLE 3.3.1-1 (CONTINUED)

ACTION STATEMENTS

c) If trip of the channel would cause a scram, place the reactor in a MODE where the limit does not apply within the next 12 hours.

DRAFT
APR 1 1985

Table 3.3.1-2 (Part 1)

INSTRUMENTATION TRIP SETPOINTS
FOR THE PLANT PROTECTIVE SYSTEM, LOOP SHUTDOWN

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1a.	Steam Pipe Rupture Under PCRV, Loop 1 (1)	≤ 8.68 VDC	≤ 8.86 VDC
1b.	Steam Pipe Rupture Under PCRV, Loop 2 (1)	≤ 8.68 VDC	≤ 8.86 VDC
1c.	Steam Pipe Rupture, North Pipe Cavity Loop 1 (1)	≤ 8.68 VDC	≤ 8.86 VDC
1d.	Steam Pipe Rupture, South Pipe Cavity Loop 1 (1)	≤ 8.68 VDC	≤ 8.86 VDC
1e.	Steam Pipe Rupture, North Pipe Cavity Loop 2 (1)	≤ 8.68 VDC	≤ 8.86 VDC
1f.	Steam Pipe Rupture, South Pipe Cavity Loop 2 (1)	≤ 8.68 VDC	≤ 8.86 VDC
2a.	High Pressure, Pipe Cavity (1)	≤ 1.3" H2O	≤ 1.3" H2O
2b.	High Temperature, Pipe Cavity (1)	≤ 125 degree F	≤ 125 degree F
2c.	High Pressure, Under PCRV (1)	≤ 1.3" H2O	≤ 1.3" H2O
2d.	High Temperature, Under PCRV (1)	≤ 125 degree F	≤ 125 degree F
3a.	Loop 1 Shutdown Logic	Not Applicable	Not Applicable
3b.	Loop 2 Shutdown Logic	Not Applicable	Not Applicable
4a.	Circulator 1A and 1B Shutdown - Loop Shutdown Logic	Not Applicable	Not Applicable

DRAFT
APR 1 1985

Table 3.3.1-2 (Part 1)

INSTRUMENTATION TRIP SETPOINTS
FOR THE PLANT PROTECTIVE SYSTEM, LOOP SHUTDOWN

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4b.	Circulator 1C and 1D Shutdown - Loop Shutdown Logic	Not Applicable	Not Applicable
5a.	Steam Generator Penetration Overpressure Loop 1	≤ 788 psig	≤ 796 psig
5b.	Steam Generator Penetration Overpressure Loop 2	≤ 788 psig	≤ 796 psig
6a.	High Reheat Header Activity, Loop 1	≤ 3.2 mr/hr Above Background	≤ 3.2 mr/hr Above Background
6b.	High Reheat Header Activity, Loop 2	≤ 3.2 mr/hr Above Background	≤ 3.2 mr/hr Above Background
7a.	Low Superheat Header Temperature, Loop 1 (m)	≥ 798 degree F	≥ 798 degree F
7b.	Low Superheat Header Temperature, Loop 2 (m)	≥ 798 degree F	≥ 798 degree F
7c.	High Differential Temperature Between Loop 1 and Loop 2 (m)	≤ 44.8 degree F	≤ 44.8 degree F
8.	Primary Coolant Moisture		
	a) High Level Monitor	----------See Table 3.3.1-1----------	
	b) Loop Monitor	----------See Table 3.3.1-1----------	

DRAFT
APR 1 1985

Table 3.3.1-2 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, LOOP SHUTDOWN

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1a.	Steam Pipe Rupture Under PCRV, Loop 1 (1)	3	2	2(n)	P,L,S/U	5
1b.	Steam Pipe Rupture Under PCRV, Loop 2 (1)	3	2	2(n)	P,L,S/U	5
1c.	Steam Pipe Rupture, North Pipe Cavity Loop 1 (1)	3	2	2	P,L,S/U	5
1d.	Steam Pipe Rupture, South Pipe Cavity Loop 1 (1)	3	2	2	P,L,S/U	5
1e.	Steam Pipe Rupture, North Pipe Cavity Loop 2 (1)	3	2	2	P,L,S/U	5
1f.	Steam Pipe Rupture, South Pipe Cavity Loop 2 (1)	3	2	2	P,L,S/U	5
2a.	High Pressure, Pipe Cavity (1)	3	2	2	P,L,S/U	5
2b.	High Temperature, Pipe Cavity (1)	3	2	2	P,L,S/U	5
2c.	High Pressure, Under PCRV (1)	3	2	2	P,L,S/U	5

DRAFT

APR 1 1985

Table 3.3.1-2 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, LOOP SHUTDOWN

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2d.	High Temperature, Under PORV (1)	3	2	2	P,L,S/U	5
3a.	Loop 1 Shutdown Logic	2	1	2	P,L,S/U, (o)	6
3b.	Loop 2 Shutdown Logic	2	1	2	P,L,S/U, (o)	6
4a.	Circulator 1A and 1B Shutdown -Loop Shutdown Logic	2	1	2 (o)	P,L,S/U,	7
4b.	Circulator 1C and 1D Shutdown -Loop Shutdown Logic	2	1	2 (o)	P,L,S/U,	7
5a.	Steam Generator Penetration Overpressure Loop 1	3	2	2	P,L,S/U	5
5b.	Steam Generator Penetration Overpressure Loop 2	3	2	2	P,L,S/U	5
6a.	High Reheat Header Activity, Loop 1	3	2	2	P,L,S/U	5
6b.	High Reheat Header Activity, Loop 2	3	2	2	P,L,S/U	5

DRAFT
APR 1 1985

Table 3.3.1-2 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, LOOP SHUTDOWN

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7a.	Low Superheat Header Temperature, Loop 1 (m)	3	2	2	P	5
7b.	Low Superheat Header Temperature, Loop 2 (m)	3	2	2	P	5
7c.	High Differential Temperature Between Loop 1 and Loop 2 (m)	3	2	2	P	5
8.	Primary Coolant Moisture					
	a) High Level Monitor	----------See Table 3.3.1-1----------				
	b) Loop Monitor	----------See Table 3.3.1-1----------				

DRAFT

APR 1 1985

Table 3.3.1-2 (CONTINUED)

TABLE NOTATION

(l) Indication of Steam Pipe Rupture Under PCRV Loop 1 (Loop 2) or Steam Pipe Rupture North Pipe Cavity Loop 1 (Loop 2) or Steam Pipe Rupture South Pipe Cavity Loop 1 (Loop 2) and indication of High Pressure-Pipe Cavity or High Temperature-Pipe Cavity or High Pressure-Under PCRV or High Temperature-Under PCRV are required for Loop 1 (Loop 2) Shutdown.

(m) Low Superheat Header Temperature Loop 1 (Loop 2) must be accompanied by High Differential Temperature between Loop 1 and Loop 2 for Loop 1 (Loop 2) Shutdown.

(n) Each channel has 2 microphones connected in parallel with an ultrasonic amplifier. For the channel to be considered OPERABLE, both microphones and the amplifier must be OPERABLE.

(o) Applicable logic for each MODE depends upon the applicable Functional Unit parameters for the MODE and which circulator, circulator drive and loop are operating or are required to be OPERABLE.

DRAFT

APR 1 1985

Table 3.3.1-2 (CONTINUED)

ACTION STATEMENTS

ACTION 5 - With the number of OPERABLE channels less than the total number of channels, restore the inoperable channel(s) to OPERABLE status within 12 hours. If the channel(s) is not restored to OPERABLE status within that 12 hours, either;

(a) Place the inoperable channel in the tripped condition unless trip of the channel would cause a loop shutdown or

(b) If trip of the channel would cause a loop shutdown, shutdown the loop or place the reactor in a MODE where the limit does not apply within the next 12 hours.

ACTION 6 - With the number of OPERABLE channels one less than the minimum channels operable requirement, restore the inoperable channel to OPERABLE status within 12 hours or shutdown the affected loop within the next 12 hours.

ACTION 7 - With the number of OPERABLE channels one less than the minimum channels operable requirement, restore the inoperable channel to OPERABLE status within 12 hours or SHUTDOWN the affected Helium Circulator (or loop) within the next 12 hours.

DRAFT
APR 1 1985

Table 3.3.1-3 (PART 1)

INSTRUMENTATION TRIP SETPOINTS FOR THE PLANT PROTECTION SYSTEM, CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1.	Manual Trip (Steam)	Not Applicable	Not Applicable
2.	Circulator Speed - High (Steam)	≤11,495 rpm	≤11,570 rpm
3.	Circulator Drain Malfunction	≥8 psid	≥8 psid
4.	Manual Trip (Water)	Not Applicable	Not Applicable
5.	Circulator Speed - High (Water)	≤8,589 rpm	≤8,670 rpm
6.	Circulator Speed - Low Programmed	≤1850 rpm Below Normal As Programmed by Feedwater Flow (4 circulators)	≤1974 rpm Below Normal As Programmed by Feedwater Flow (4 circulators) per Figure 3.3.1-1
7a.	Loop 1, Fixed Feedwater Flow - Low (Both Circulators)	≥177,500 lb/hr (15.4% of normal Full Load)	≥171,750 lb/hr (14.9% of normal Full Load)
7b.	Loop 2, Fixed Feedwater Flow - Low (Both Circulators)	≥177,500 lb/hr (15.4% of normal Full Load)	≥171,750 lb/hr (14.9% of normal Full Load)

DRAFT
APR 1 1985

Table 3.3.1-3 (PART 1)

INSTRUMENTATION TRIP SETPOINTS FOR THE PLANT PROTECTION SYSTEM, CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
8a.	Loop 1, Programmed Feedwater Flow - Low (Both Circulators)	≤211,000 lb/hr (18.3%) Below normal as programmed by Circulator Speed. (4 circulators)	≤230,530 lb/hr (20%) Below normal as programmed by Circulator Speed. (4 circulators) per Figure 3.3.1-1
8b.	Loop 1, Programmed Feedwater Flow - Low (One Circulator)	≤211,000 lb/hr (18.3%) Below normal as programmed by Circulator Speed. (2 circulators)	≤230,530 lb/hr (20%) Below normal as programmed by Circulator Speed. (2 circulators) per Figure 3.3.1-2
9a.	Loop 2, Programmed Feedwater Flow - Low (Both Circulators)	≤211,000 lb/hr (18.3%) Below normal as programmed by Circulator Speed. (4 circulators)	≤230,530 lb/hr (20%) Below normal as programmed by Circulator Speed. (4 circulators) per Figure 3.3.1-1
9b.	Loop 2, Programmed Feedwater Flow - Low (One Circulator)	≤211,000 lb/hr (18.3%) Below normal as programmed by Circulator Speed. (2 circulators)	≤230,530 lb/hr (20%) Below normal as programmed by Circulator Speed (2 circulators) per Figure 3.3.1-2

DRAFT

APR 1 1985

Table 3.3.1-3 (PART 1)

INSTRUMENTATION TRIP SETPOINTS FOR THE PLANT PROTECTION SYSTEM, CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
10a.	Circulator Seal Malfunction-Low	≥-6" H20,	≥-6" H20,
10b.	Circulator Seal Malfunction-High	≤+75.5" H20	≤+75.5" H20
11.	Loss of Circulator Bearing Water	≥459 psid	≥459 psid
12.	Circulator Penetration Overpressure	≤786 psig	≤796 psig

DRAFT
APR 1 1985

Table 3.3.1-3 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Manual Trip (Steam)	1	1	1	All Modes(r)	8
2.	Circulator Speed-High (Steam)	3	2	2	All Modes(r)	8
3.	Circulator Drain Mal-function	3	2	2	P,L,S/U, S/D(s)	8
4.	Manual Trip (Water)	1	1	1	All Modes(r)	8
5.	Circulator Speed-High (Water)	3	2	2	All Modes(r)	9
6.	Circulator Speed-Low Programmed	3	2	2	P	8
7a.	Loop 1, Fixed Feedwater Flow-Low (Both Circulators)	3	2	2	P	8
7b.	Loop 2, Fixed Feedwater Flow-Low (Both Circulators)	3	2	2	P	8
8a.	Loop 1, Programmed Feedwater Flow-Low (Both Circulators)	3	2	2	P	8

DRAFT
APR 1 1985

Table 3.3.1-3 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8b.	Loop 1, Programmed Feedwater Flow-Low (One Circulator)	3	2	2	P	8
9a.	Loop 2, Programmed Feedwater Flow-Low (Both Circulators)	3	2	2	P	8
9b.	Loop 2, Programmed Feedwater Flow-Low (One Circulator)	3	2	2	P	8
10a.	Circulator Seal Mal-function-Low	3	2	2(t)	All Modes(r)	8
10b.	Circulator Seal Mal-function-High	3	2	2(t)	All Modes(r)	8
11.	Loss of Circulator Bearing Water	3	2	2	All Modes(r)	8
12.	Circulator Penetration Overpressure	3	2	2	All Modes(r)	8

DRAFT

APR 1 1985

Table 3.3.1-3 (Continued)

TABLE NOTATION

(r) Applicable in MODES where the circulator is required to be OPERABLE per Specification 3.5.1.

(s) Applicable when the helium circulator(s) is being driven by steam.

(t) Circulator seal malfunction channels can be bypassed when the opposite loop is shutdown or when the circulator seal malfunction is tripped on the other circulator in the same loop.

DRAFT

APR 1 1985

Table 3.3.1-3 (CONTINUED)

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels less than the total number of channels, restore the inoperable channel(s) to OPERABLE status within 12 hours. If the channel(s) is not restored to OPERABLE status within that 12 hours, either;

(a) Place the inoperable channel in the tripped condition unless trip of the channel would cause a helium circulator (or loop) shutdown or

(b) If trip of the channel would cause a helium circulator (or loop) shutdown, shutdown the affected helium circulator (or loop) or place the reactor in a MODE where the limit does not apply within the next 12 hours.

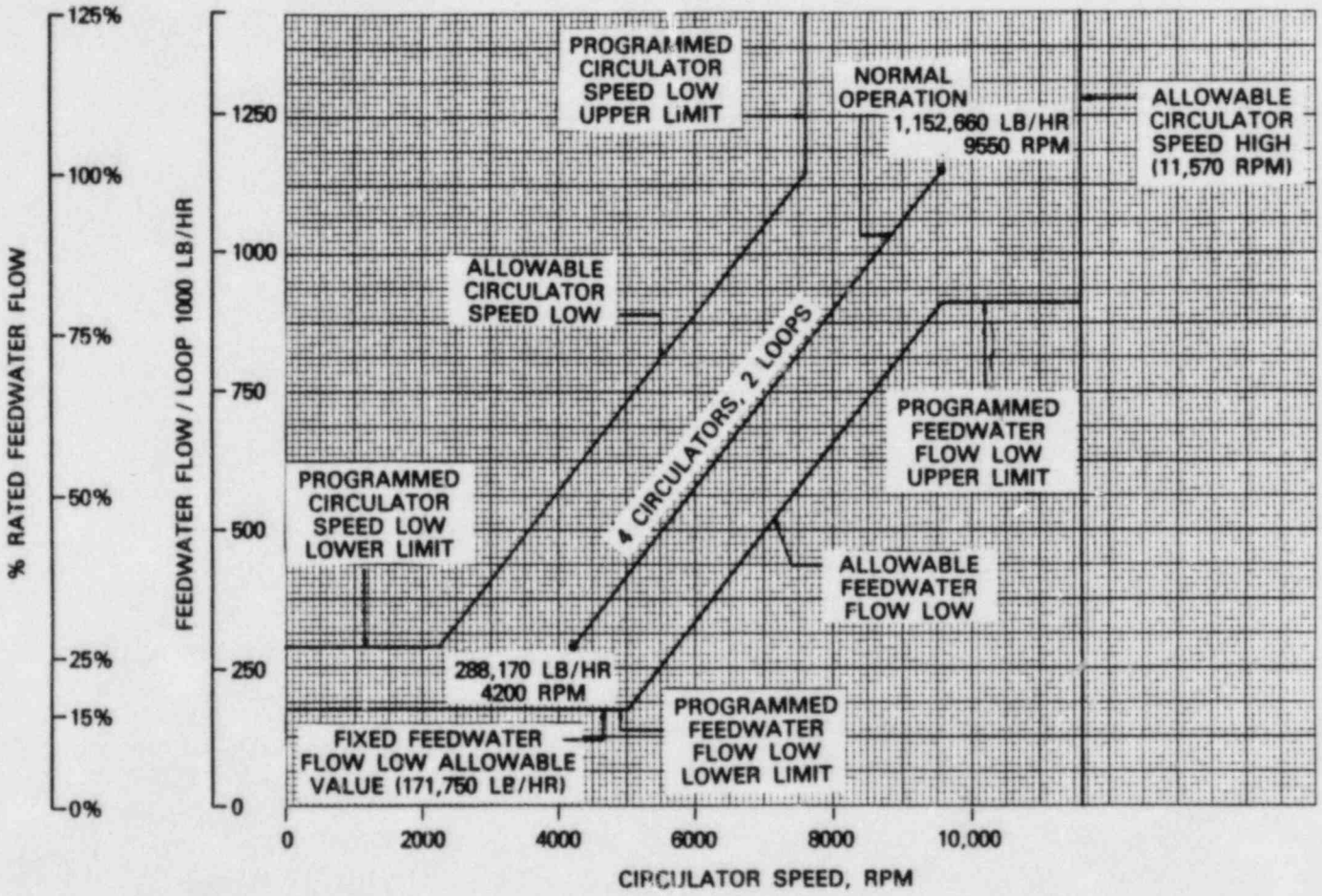
ACTION 9 - With the number of OPERABLE channels less than the total number of channels, restore the inoperable channel(s) to OPERABLE status within 12 hours. If the channel(s) is not restored to OPERABLE status within that 12 hours, either;

(a) Place the inoperable channel in the tripped condition unless trip of the channel would cause a helium circulator (or loop) shutdown or

(b) If trip of the channel would cause a helium circulator (or loop) shutdown to occur, shutdown the affected helium circulator (or loop), or declare the water turbine drive inoperable and comply with Specification 3.5.1 and

(c) If the total number of channels is not maintained OPERABLE on at least one circulator per loop, the reactor shall be reduced to 50% of RATED THERMAL POWER within 12 hours of discovery.

DRAFT
APR 1 1985

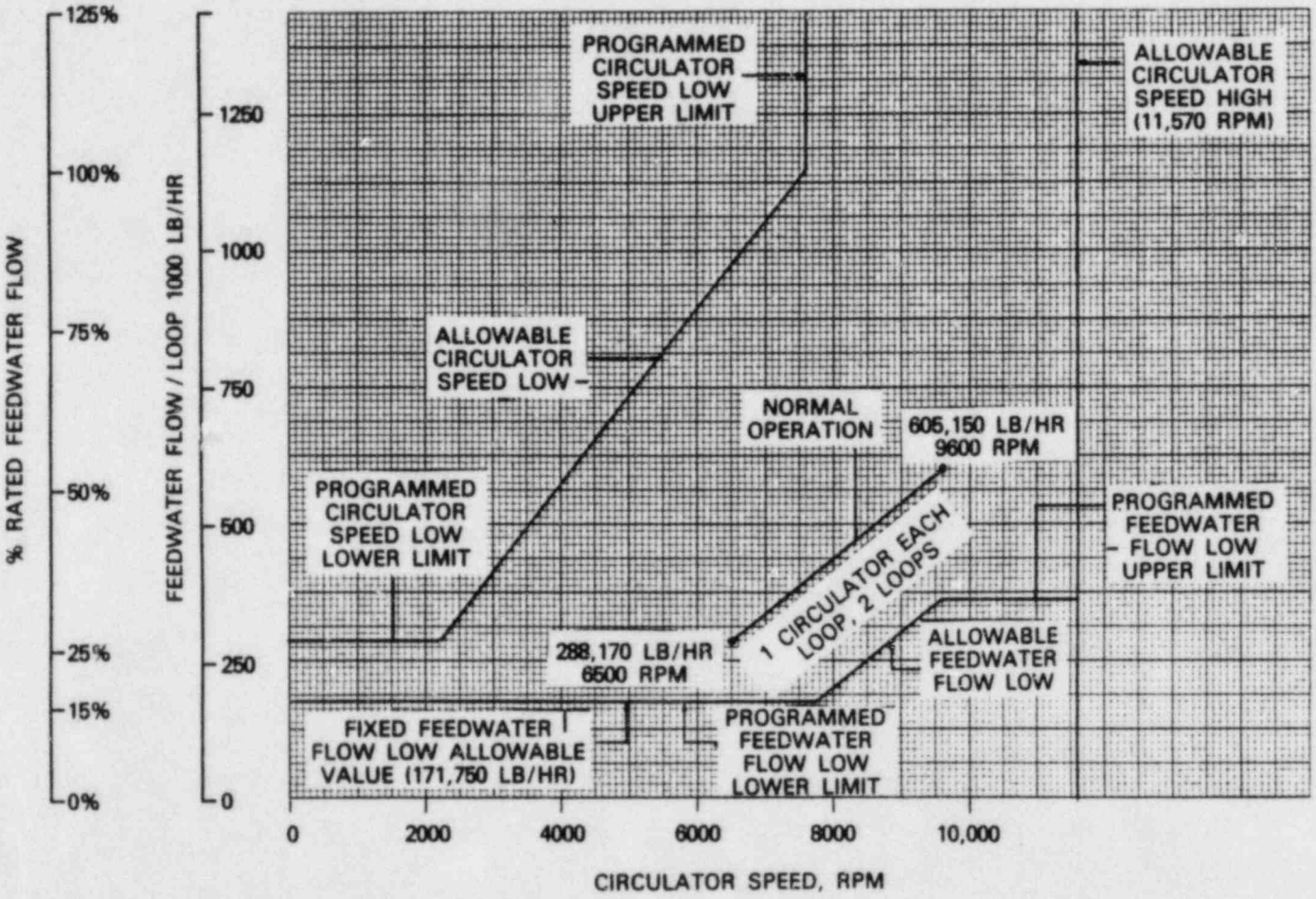


FEEDWATER FLOW vs. CIRCULATOR SPEED
FOR OPERATION IN THE POWER MODE

(4 CIRCULATORS)

Figure 3.3.1–1

DRAFT
APR 1 1985



FEEDWATER FLOW vs. CIRCULATOR SPEED
FOR OPERATION IN THE POWER MODE

(2 CIRCULATORS)

Figure 3.3.1–2

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APR 1 1985

Table 3.3.1-4 (Part 1)

INSTRUMENTATION TRIP SETPOINTS FOR THE PLANT PROTECTIVE SYSTEM, ROD WITHDRAWAL PROHIBIT (RWP)

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1.	Start-up channel-Low Count rate (Channels 1 and 2)	≥ 4.2 cps	≥ 3.2 cps
2a.	Linear Channel-5% RWP* (Channels 3, 4 and 5)	≤ 5% (x)	≤ 5%
2b.	Linear Channel-5% RWP* (Channels 6, 7 and 8)	≤ 5% (x)	≤ 5%
3a.	Linear Channel-30% RWP* (Channels 3, 4 and 5)	≤ 30% (y)	≤ 30%
3b.	Linear Channel-30% RWP* (Channels 6, 7 and 8)	≤ 30% (y)	≤ 30%
4a.	Wide Range Channel Rate of Change - High (Channels 3, 4 and 5)	≤ 1.5 dpm	≤ 2 dpm
4b.	Start-up Channels Rate of Change - High (Channels 1 and 2)	≤ 1.5 dpm	≤ 2 dpm
5a.	Linear Channel-High power RWP (Channels 3, 4 and 5)	Varies as a Function of Indicated Thermal Power (z)	Varies as a Function of Indicated Thermal Power (z)

* % of RATED THERMAL POWER

DRAFT
APR 1 1985

Table 3.3.1-4 (Part 1)

INSTRUMENTATION TRIP SETPOINTS FOR THE PLANT PROTECTIVE SYSTEM, FOR ROD WITHDRAWAL PROHIBIT (RWP)

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
5b.	Linear Channel-High power RWP (Channels 6, 7 and 8)	Varies as a Function of Indicated Thermal Power (z)	Varies as a Function of Indicated Thermal Power (z)
6.	Multiple Rod Pair Withdrawal	Not Applicable	Not Applicable

DRAFT

APR 1 1985

Table 3.3.1-4 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR THE PLANT PROTECTIVE SYSTEM, ROD WITHDRAWAL PROHIBIT (RWP)

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Start-up Channel-Low Count Rate (Channels 1 and 2)	2	1	2	S/U,R(aa) (bb)	10
2a.	Linear Channel-5% RWP (Channels 3, 4 and 5)	3	2	2	S/U(x)	11
2b.	Linear Channel-5% RWP (Channels 6, 7 and 8)	3	2	2	S/U(x)	11
3a.	Linear Channel-30% RWP (Channels 3, 4 and 5)	3	2	2	L(y)	11
3b.	Linear Channel-30% RWP (Channels 6, 7 and 8)	3	2	2	L(y)	11
4a.	Wide Range Channel Rate of Change-High (Channels 3, 4 and 5)	3	2	2	S/U	11
4b.	Start-up Channels Rate of Change-High (Channels 1 and 2)	2	1	2	S/U(bb)	10

DRAFT

APR 1 1985

Table 3.3.1-4 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR THE PLANT PROTECTIVE SYSTEM, ROD WITHDRAWAL PROHIBIT (RWP)

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5a.	Linear Channel-High Power RWP (Channels 3, 4 and 5)	3	2	2	P,L,S/U	11
5b.	Linear Channel-High Power RWP (Channels 6, 7 and 8)	3	2	2	P,L,S/U	11
6.	Multiple Rod Pair Withdrawal	----------Not Applicable----------				

DRAFT

APR 1 1985

Table 3.3.1-4 (Continued)

TABLE NOTATION

(x) The Low power RWP bistable automatically resets at approximately 4% of RATED THERMAL POWER after reactor power is decreased from greater than 5% of RATED THERMAL POWER.

(y) The Power range RWP bistables automatically reset at approximately 10% of RATED THERMAL POWER after reactor power is decreased from greater than 30% of RATED THERMAL POWER. The RWP may be manually reset between approximately 10% and 30% of RATED THERMAL POWER.

(z) Curves specifying the Linear Channel High Neutron Flux trip setpoint limit and allowable value as a function of indicated power level, which account for neutron detector decalibration, shall be established for each fuel cycle. The neutron detector decalibration curves shall be approved by the NFSC prior to each fuel cycle. The detector decalibration curves approved by NFSC shall be forwarded within 30 days of approval to the Regional Administrator; Region IV, Nuclear Regulatory Commission. See Tables 3.3.1-1 and 2.2.1-1 for related limits.

(aa) The applicability for REFUELING is anytime the reactor scram is reset and the control rod drive system is capable of rod withdrawal.

(bb) The Start-up channel may be disabled above E-03% of RATED THERMAL POWER.

DRAFT

APR 1 1985

Table 3.3.1-4 (CONTINUED)

ACTION STATEMENTS

ACTION 10 - With the number of OPERABLE channels one less than the minimum channels operable requirement;

(a) Suspend all operations involving control rod withdrawl or fuel manipulation in the core and terminate incore maintenance per Specification 3.9.1 and

(b) Either restore the inoperable channel to OPERABLE status within 12 hours, or actuate the Rod Withdrawal Prohibit, or be in at least STARTUP within the next 12 hours and SHUTDOWN in the next 12 hours.

ACTION 11 - With the number of OPERABLE channels less than the total number of channels, restore the inoperable channel(s) to OPERABLE status within 12 hours or suspend all operations involving control rod withdrawal or fuel manipulation in the core and terminate incore maintenance per Specification 3.9.1.

DRAFT
APR 1 1985

Table 3.3.1-5 (Part 1)

TRIP SETPOINTS FOR 480V AC ESSENTIAL BUS UNDERVOLTAGE PROTECTION

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	TIME DELAY	ALLOWABLE VALUE
1.	Plant Electrical System - Loss (Scram)	≥ 278V See also Table 3.3.1-1	28.5 - 31.5 seconds	≥ 278V ≤ 35 seconds
2.	Degraded Voltage	396V - 436V	115 - 125 seconds	Not Applicable
3.	Loss of Voltage-Automatic Throw Over (ATO)	361V - 383V	CV-2 Relay Setting of Time Dial 5	Not Applicable
4.	Loss of Voltage-D.G. Start, Load Shed and Load Sequence	318V - 338V	CV-2 Relay Setting of Time Dial 6	Not Applicable

DRAFT
APR 1 1985

Table 3.3.1-5 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR 480 VAC ESSENTIAL BUS UNDERVOLTAGE PROTECTION

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Plant Electrical System - Loss (Scram)	----------See Table 3.3.1-1----------				
2.	Degraded Voltage	3	2	2(dd)	All Modes	12
3.	Loss of Voltage-Automatic Throw-over (ATO)	3	2	2(dd)	All Modes	12
4.	Loss of Voltage-D.G. Start, Load Shed and Load Sequence	3	2	2(ee)	All Modes	12

Table 3.3.1-5 (Continued)

DRAFT
APR 1 1985

TABLE NOTATION

(dd) A channel consists of one relay monitoring a single phase of a 480 VAC essential bus. The protective action occurs when two of three relays are activated.

(ee) One channel consists of three undervoltage relays each monitoring a single phase of a 480 VAC essential bus. A channel trip will occur when two of the three undervoltage relays comprising that channel operate after a preset time delay indicating loss of bus voltage. Initiation of a scram requires two of the three undervoltage relays on two of the three 480 VAC essential buses to operate.

DRAFT
APR 1 1985

Table 3.3.1-5 (CONTINUED)

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels less than the total number of channels, restore the inoperable channel(s) to OPERABLE status within 12 hours. If the channel(s) is not restored to OPERABLE status within that 12 hours, either;

(a) Place the inoperable channel in the tripped condition unless trip of the channel would cause the protective action to occur or

(b) If trip of the channel would cause the protective action to occur, declare the affected 480 VAC essential bus inoperable and comply with the requirements of Specification 3.8.3.

DRAFT
APR 1 1985

Table 4.3.1-1

REACTOR SCRAM SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
1a.	Manual Scram (Control Room)			P(a)		P,L,S/U, R(b)
1b.	Manual Scram (Outside Control Room)			P,M(c)		P,L,S/U, R(b)
2a.	Start-up Channel-High	D	R(d)	P(e)		R(b)
2b.	Wide Range Channel Rate of Change-High	D	M(f),R(g)	P(e)		S/U
3a.	Linear Channel-High Channels 3,4,5	D	D(f),R(g)	M(e)		P,L,S/U
3b.	Linear Channel-High Channels 6,7,8	D	D(f),R(g)	M(e)		P,L,S/U
4.	Primary Coolant Moisture					
	a) High Level Monitor	D(h), D(i)	R(j)	M(k)	M(l)	P,L,S/U
	b) Loop Monitor	D(h), D(i)	R(j)	M(k)(m)	M(n)	P,L,S/U, (a7)
5.	Reheat Steam Temperature -High	D(o)	R(p)(g)	M(q)		P,L,S/U
6a.	Primary Coolant Pressure -Programmed Low	D(r)	R(s)(g)	M(q)		P

DRAFT

APR 1 1985

Table 4.3.1-1 (Continued)

REACTOR SCRAM SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
6b.	Circulator Inlet Temperature -Programmed High	D(t)	R(p)(g)	M(q)		P
7.	Primary Coolant Pressure -Programmed High	D(r)	R(s)(g)	M(q)	M(u)	P,L,S/U
8.	Hot Reheat Header Pressure -Low		R(v)	M(w)		P
9.	Main Steam Pressure -Low		R(v)	M(w)		P
10.	Plant Electrical System-Loss			M(x)		P,L,S/U
11.	Two Loop Trouble Scram Logic			M(y)	R(z)	P,L,S/U
12.	High Reactor Building Temperature (Pipe Cavity)	D	R(p)	M(q)		P,L,S/U

DRAFT
APR 1 1985

Table 4.3.1-1 (Continued)

TABLE NOTATION

(a) Manually trip the system.

(b) The applicability for REFUELING MODE is anytime the reactor scram is reset and the control rod drive system is capable of rod withdrawal.

(c) Manually trip each channel.

(d) Internal test signal shall be checked and calibrated to assure that its output is in accordance with the design requirements. This shall be done after completing the external test signal procedure by checking the output indication when turning the internal test signal switch.

(e) An internal test signal is applied to verify trip setpoints and alarms.

(f) The channel is adjusted to agree with the heat balance calculation when the reactor is critical.

(g) An internal test signal is applied to adjust trip setpoints and indicators.

(h) Verification of sample flow of OPERABLE moisture monitors per note (f) of Table Notation for Table 3.3.1-1.

(i) Compare two or more mirror temperature indicators.

(j) Inject moisture laden gas into sample lines.

(k) Verify that each of the OPERABLE moisture monitors sample flow alarms are OPERABLE.

(l) Trip one high level and one low level channel, pulse another low level channel.

(m) Trip each channel and verify proper indications.

(n) Trip each channel, pulse test other loop to check loop identification.

(o) Compare the averaged themocouple channel input indications.

(p) Compare each thermocouple output to an NBS traceable standard.

DRAFT

APR 1 1985

Table 4.3.1-1 (Continued)

TABLE NOTATION

(q) Trip channel, verify alarms and indications. Internal test signal to verify trip setpoints and alarms.

(r) Comparison of three programmed channel indicators.

(s) Known pressure applied to sensor. Internal test signal to verify trip setpoints and indicators.

(t) Comparison of six separate temperature indicators.

(u) Pulse test one channel with another channel tripped and verify proper indications, both channels.

(v) Known pressure applied at sensor to adjust trip setpoints.

(w) Reduce pressure at sensor to trip channel; verify alarms and indications.

(x) Trip each channel by applying simulated loss of voltage signal, verify alarms and indications.

(y) Special test module used to trip channel by energizing each of four appropriate pairs of two-loop trouble relays.

(z) Trip logic to cause two loop trouble scram.

(a7) Permissible bypass conditions

1. Any circulator buffer seal malfunction

2. Either loop hot reheat header high activity

DRAFT

APR 1 1985

Table 4.3.1-2

LOOP SHUTDOWN SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
1a.	Steam Pipe Rupture Under PCRV, Loop 1	D	R(aa)	M(bb)(cc)	M(dd)(ee)	P,L,S/U
1b.	Steam Pipe Rupture Under PCRV, Loop 2	D	R(aa)	M(bb)(cc)	M(dd)(ee)	P,L,S/U
1c.	Steam Pipe Rupture, North Pipe Cavity Loop 1	D	R(aa)	M(bb)(cc)	M(dd)(ee)	P,L,S/U
1d.	Steam Pipe Rupture, South Pipe Cavity Loop 1	D	R(aa)	M(bb)(cc)	M(dd)(ee)	P,L,S/U
1e.	Steam Pipe Rupture, North Pipe Cavity Loop 2	D	R(aa)	M(bb)(cc)	M(dd)(ee)	P,L,S/U
1f.	Steam Pipe Rupture, South Pipe Cavity Loop 2	D	R(aa)	M(bb)(cc)	M(dd)(ee)	P,L,S/U
2a.	High Pressure, Pipe Cavity		R(v)	M(ff)		P,L,S/U
2b.	High Temperature, Pipe Cavity		R(gg)	M(hh)		P,L,S/U
2c.	High Pressure, Under PCRV		R(v)	M(ff)		P,L,S/U
2d.	High Temperature, Under PCRV		R(gg)	M(hh)		P,L,S/U

DRAFT

APR 1 1985

Table 4.3.1-2 (Continued)

LOOP SHUTDOWN SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
3a.	Loop 1 Shutdown Logic				R(ii)	P,L,S/U, (jj)
3b.	Loop 2 Shutdown Logic				R(ii)	P,L,S/U, (jj)
4a.	Circulator 1A and 1B Shutdown -Loop Shutdown Logic				M(kk)	P,L,S/U, (jj)
4b.	Circulator 1C and 1D Shutdown -Loop Shutdown Logic				M(kk)	P,L,S/U, (jj)
5a.	Steam Generator Penetration Overpressure Loop 1		R(v)	M(ff)	M(ll)	P,L,S/U
5b.	Steam Generator Penetration Overpressure Loop 2		R(v)	M(ff)	M(ll)	P,L,S/U
6a.	High Reheat Header Activity, Loop 1	D(mm)	R(nn)(g)		M(ll)	P,L,S/U
6b.	High Reheat Header Activity, Loop 2	D(mm)	R(nn)(g)		M(ll)	P,L,S/U
7a.	Low Superheat Header Tempera-ture, Loop 1	D(mm)	R(p)(g)		M(oo)	P

DRAFT
APR 1 1985

Table 4.3.1-2 (Continued)

LOOP SHUTDOWN SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
7b.	Low Superheat Header Temperature, Loop 2	D(mm)	R(p)(g)		M(oo)	P
7c.	High Differential Temperature Between Loop 1 and Loop 2	D(mm)	R(p)(g)		M(oo)	P
8.	Primary Coolant Moisture					
	a) High Level	----------See Table 4.3.1-1----------				
	b) Loop Monitor	----------See Table 4.3.1-1----------				

DRAFT
APR 1 1985

Table 4.3.1-2 (Continued)

TABLE NOTATION

(aa) Known sound reference applied at sensor to adjust channel gain.

(bb) Internal sign to adjust ultrasonic trip setpoints.

(cc) Trip test signal solenoid valves to verify loop integrity.

(dd) Pulse test one temperature channel with another temperature channel tripped, while simultaneously having two ultrasonic channels tripped.

(ee) Pulse test one pressure channel with another pressure channel tripped, while simultaneously having two ultransonic channels tripped.

(v) Known pressure applied at sensor to adjust trip setpoints.

(ff) Pressure switch actuated by pressure applied at sensor.

(gg) Known temperature applied at sensor to adjust trip setpoints.

(hh) Temperature switch actuated by heat applied at sensor.

(ii) Trip both circulators to test loop shutdown.

(jj) Applicable logic for each MODE depends upon the applicable Functional Unit parameters for the MODE and which circulator, circulator drive, and loop are operating or are required to be OPERABLE.

(kk) Pulse test and verify proper indications.

(ll) Pulse test each channel with another channel tripped and verify proper indications.

(mm) Comparison of three separate indicators in each loop.

(nn) Expose sensor to known radiation source for calbration.

(f) Compare each thermocouple output to an NBS traceable standard.

(g) An internal test signal is applied to adjust trip setpoints and indicators.

(oo) Pulse test one temperature channel and one temperature differential channel with one temperature and one temperature differential channel tripped.

DRAFT
APR 1 1985

Table 4.3.1-3

CIRCULATOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
1.	Manual Trip (Steam)			R(pp)		All Modes (qq)
2.	Circulator Speed-High (Steam)	D	R(rr)	M(ss)	M(tt)	All Modes (qq)
3.	Circulator Drain Mal-function	D	R(v)		M(tt)	All Modes (uu)
4.	Manual Trip (Water)			R(vv)		All Modes (qq)
5.	Circulator Speed-High (Water)	D	R(rr)	M(ss)	M(tt)	All Modes (qq)
6.	Circulator Speed-Low Programmed	D	R(rr)	M(ss)	M(tt)	P
7a.	Loop 1, Fixed Feedwater Flow-Low (Both Circulators)	D	R(g)(ww)	M(ss)	M(tt)	P
7b.	Loop 2, Fixed Feedwater FLow-Low (Both Circulators)	D	R(g)(ww)	M(ss)	M(tt)	P
8a.	Loop 1, Programmed Feedwater Flow-Low (Both Circulators)	D	R(g)(ww)	M(ss)	M(tt)	P

DRAFT
APR 1 1985

Table 4.3.1-3 (Continued)
CIRCULATOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
8b.	Loop 1, Programmed Feedwater Flow-Low (One Circulator)	D	R(g)(ww)	M(ss)	M(tt)	P
9a.	Loop 2, Programmed Feedwater Flow-Low (Both Circulators)	D	R(g)(ww)	M(ss)	M(tt)	P
9b.	Loop 2, Programmed Feedwater Flow-Low (One Circulator)	D	R(g)(ww)	M(ss)	M(tt)	P
10a.	Circulator Seal Mal-function-Low	D	R(g)(xx)	M(ss)	M(tt)	All Modes (qq)
10b.	Circulator Seal Mal-function-High	D	R(g)(xx)	M(ss)	M(tt)	All Modes (qq)
11.	Loss of Circulator Bearing Water	D	R(v)		M(tt)	All Modes (qq)
12.	Circulator Penetration Overpressure		R(v)	M(ff)	M(tt)	All Modes (qq)

DRAFT

APR 1 1985

Table 4.3.1-3 (Continued)

TABLE NOTATION

(pp) Trip steam turbine drives. Verify water turbine automatic start.

(qq) Applicable in MODES where the circulator is required to be OPERABLE per Specification 3.5.1.

(rr) A known pulse frequency is applied to adjust trip setpoints and indicators.

(ss) An internal test signal is applied to verify trip setpoints and indicators.

(tt) Pulse test one channel with another channel tripped and verify proper indications.

(v) Known pressure applied at sensor to adjust trip setpoints.

(uu) Applicability in SHUTDOWN is when the helium circulator(s) is being driven by steam.

(vv) Trip water turbine drives and verify proper indications.

(g) An internal test signal is applied to adjust trip setpoints and indicators.

(ww) A known differential pressure is applied at the flow transmitter.

(xx) A known differential pressure is applied at the sensor.

(ff) Pressure switch actuated by pressure applied at sensor.

DRAFT
APR 1 1985

Table 4.3.1-4

ROD WITHDRAWAL PROHIBIT INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
1.	Start-up Channel-Low Count Rate (Channels 1 and 2)	D	R(d)	P(e)		S/U,R(b) (yy)
2a.	Linear Channel 5% RWP (Channels 3, 4 and 5)	D	D(f), R(g)	M(e)		S/U(a1)
2b.	Linear Channel 5% RWP (Channels 6, 7 and 8)	D	D(f), R(g)	M(e)		S/U(a1)
3a.	Linear Channel 30% RWP (Channels 3, 4 and 5)	D	D(f), R(g)	M(e)		L(a2)
3b.	Linear Channel 30% RWP (Channels 6, 7 and 8)	D	D(f), R(g)	M(e)		L(a2)
4a.	Wide Range Channel Rate of Change-High (Channels 3, 4 and 5)	D	M(f), R(g)	P(e)		S/U
4b.	Start-up Channels Rate of Change-High (Channels 1 and 2)	D	R(d)	P(e)		S/U(yy)

DRAFT

APR 1 1985

Table 4.3.1-4 (Continued)

ROD WITHDRAWAL PROHIBIT INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRA-TION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
5a.	Linear Channel-High Power RWP (Channels 3, 4 and 5)	D	D(f), R(g)	M(e)		P,L,S/U
5b.	Linear Channel-High Power RWP (Channels 6, 7 and 8)	D	D(f), R(g)	M(e)		P,L,S/U
6.	Multiple Rod Pair Withdrawal			P(a3) R(a4)		All Modes (b)

DRAFT

APR 1 1985

Table 4.3.1-4 (Continued)

TABLE NOTATION

(d) Internal test signal shall be checked and calibrated to assure that its output is in accordance with the design requirements. This shall be done after completing the external test signal procedure by checking the output indication when turning the internal test signal switch.

(e) An internal test signal is applied to verify trip setpoints and alarms.

(b) The applicability for REFUELING is anytime the reactor scram is reset and the control rod drive system is capable of rod withdrawal.

(yy) The start-up channel may be disabled above E-03% of RATED THERMAL POWER.

(f) The channel is adjusted to agree with the heat balance calculation when the reactor is critical.

(g) An internal test signal is applied to adjust trip setpoints and indicators.

(a1) The Low Power RWP bistable automatically resets at approximately 4% of RATED THERMAL POWER after reactor power is decreased from greater than 5% of RATED THERMAL POWER.

(a2) The Power Range RWP bistables automatically reset at approximately 10% of RATED THERMAL POWER after reactor power is decreased from greater than 30% of RATED THERMAL POWER. The RWP may be manually reset between approximately 10% and 30% of RATED THERMAL POWER.

(a3) Attempt a two rod pair withdrawal. Check for actuation of prhibit.

(a4) A current is simulated through the sensor to verify trip setpoint and alarms.

DRAFT
APR 1 1985

Table 4.3.1-5

480V AC ESSENTIAL BUS UNDERVOLTAGE PROTECTION

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABLE MODES
1.	Plant Electrical System -Loss (Scram)	----------See Table 4.3.1-1----------				
2.	Degraded Voltage		R(a5)	M(a6)		All Modes
3.	Loss of Voltage-D.G. Start, Load Shed and Load Sequence		R(a5)	M(a6)		All Modes
4.	Loss of Voltage-Automatic Throwover (ATO)		R(a5)	M(a6)		All Modes

DRAFT

APR 1 1985

Table 4.3.1-5 (Continued)

TABLE NOTATION

(a5) Trip each channel by applying a simulated voltage signal (decreased or loss of); Adjust trip setpoints and verify alarms and indications.

(a6) Simulate loss of voltage on one channel and verify alarms, indications, relay actuation, and time delays.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1

The PLANT PROTECTIVE SYSTEM automatically initiates protective functions to prevent established limits from being exceeded. In addition, other protective instrumentation is provided to initiate action which mitigates the consequences of accidents. This specification provides the limiting conditions for operation necessary to preserve the effectiveness of these instrument systems.

The trip setpoints are included in this section of the Specification. The bases for these settings are briefly discussed below. Additional discussions pertaining to the scram, loop shutdown and circulator trip inputs may be found in Sections 7.1.2.3, 7.1.2.4, and 7.1.2.6, respectively, of the FSAR. High moisture instrumentation is discussed in FSAR Section 7.3.2.

a) Scram Inputs

The simultaneous insertion of the control rods will be initiated by the following conditions:

Manual Scram

A Manual Scram is provided to give the operator means for emergency shutdown of the reactor independent of the automatic reactor protective system. The Reactor Mode Switch (RMS) in the "off" position also causes a manual scram.

Start-up Channel - High Count Rate

High start-up count rate is provided as a scram for use during fuel loading and preoperational testing or other low-power operations.

Linear Channel - High (Neutron Flux)

See Specification 2.2

Wide Range Channel - Rate of Change-High

High rate of change of neutron flux is used as a scram input during plant start-up and results in additional protection to the Linear Channel - High scram in case of accidental control rod withdrawal. The trip setpoint is selected to be above the operating rate of flux change. This scram trip setpoint is active only when the Interlock Sequence Switch is in Start-up position.

Primary Coolant Moisture - High

See Specification 2.2.

Reheat Steam Temperature - High

See Specification 2.2

DRAFT

APR 1 1985

Primary Coolant Pressure - Programmed Low

See Specification 2.2.

Primary Coolant Pressure - Programmed High

See Specification 2.2.

Hot Reheat Header Pressure - Low

Low reheat steam pressure is an indication of either a cold reheat steam line or a hot reheat steam line rupture in a section of line common to both loops. Loss of the cold reheat steam line results in loss of the steam supply to the circulators which necessitates plant shutdown. The direct scram in this case precedes a scram resulting from the two-loop trouble. The loss of either steam line results in loss of plant generation output, and a reactor scram is appropriate in this situation. The trip setpoint is selected to be below normal operating and transient levels, which vary over a wide range.

Main Steam Pressure - Low

Low main steam pressure is an indication of main steam line rupture or loss of feedwater flow. Immediate shutdown of the reactor is appropriate in such a situation. In addition, the superheater outlet stop check values are automatically closed to reroute main steam to the flash tank (through the individual loop bypass valves and desuperheaters). This is necessary for the continued operation of the helium circulators on steam. The trip setpoint is selected to be below normal operating levels and system transients.

Plant Electrical System Power-Loss

Loss of plant electrical system power requires a scram to prevent any power-to-flow mismatches from occurring. A preset time delay is provided following a power loss before the scram is initiated to allow an emergency diesel generator to start. If it does start, the scram is avoided.

Two-Loop Trouble Scram Logic

Operation on one loop at a maximum of about 50% power may continue following the shutdown of the other loop (unless preceded by scram as in the case of high moisture). Onset of trouble in the remaining loop (two-loop trouble) results in a scram. Trouble is defined as a signal which normally initiates a loop shutdown. Similarly, simultaneous shutdown signals to both loops result in shutdown of one of the two loops only, and a reactor scram.

High Reactor Building Temperature, Pipe Cavity

High temperature in the pipe cavity would indicate the presence of an undetected steam leak or the failure of the steam pipe rupture detection system to determine the loop in which the leak had occurred and to shut the faulty loop down.

DRAFT

APR 1 1985

The trip setpoint has been set above the temperature that would be expected to occur in the pipe cavity if the steam leak were detected and the faulty loop shutdown for all steam leaks except those of major proportion such as that due to an offset rupture of one of the steam lines.

An undetected steam leak or pipe rupture under the PCRV within the support ring would also be detectable in the pipe cavity, therefore only one set of sensors and logic is required to monitor both areas.

b) Loop Shutdown Inputs

The following loop shutdown inputs are provided primarily for equipment protection and are not relied upon to protect Safety Limits. Malfunction of these items could prevent a scram due to loss of the two loop trouble scram input.

Steam Pipe Rupture In The Reactor Building

The purpose of the ultrasonic detectors is to identify the specific secondary coolant loop within the reactor building containing a pipe rupture. Ultrasonic noise caused by escaping steam in conjuction with a pressure or temperature rise will cause the appropriate loop to shut down.

The trip of the ultrasonic detection system is set at a level which corresponds to 8.68 VDC output from the ultrasonic amplifier. The pressure and temperature trips are set above normal operating building pressure and temperature levels.

High Pressure - Pipe Cavity

The trip setpoint is above normal reactor building pressure of 0.25" w.g. but below the pressure of about 3" w.g. at which the reactor building louvers open to relieve any overpressure condition.

High Temperature - Pipe Cavity

The trip setpoint is established to be above the normal ambient temperature in the pipe cavity, and low enough to assure a fast response to steam pipe ruptures in the pipe cavity.

High Pressure, Under PCRV

The trip setpoint is above normal reactor building pressure of 0.25" w.g. but below the pressure of 3" w.g. at which the reactor building louvers open to relieve any overpressure condition.

High Temperature, Under PCRV

The trip setpoint is established to be above the normal temperature inside the PCRV support ring to preclude spurious trips. The ambient temperature under the PCRV is normally higher than that in the pipe cavity. Conversely, the trip setpoint is low enough not to preclude a fast response in the event of a steam pipe rupture.

DRAFT

APR 1 1985

Shutdown of Both Circulators (Loop Shutdown Logic)

Shutdown of both circulators is a loop shutdown input which is necessary to insure proper action of the reactor protective (scram) system through the two-loop trouble scram in the event of the loss of all circulators and low feedwater flow.

Steam Generator Penetration - Overpressure (Loop 1/Loop 2)

Steam generator penetration overpressure is indicative of a pipe rupture within the penetration. A loop shutdown is appropriate for such an accident, and the helium pressurizing line to the penetration is closed to prevent moisture backflow to the purified helium system. The penetration overpressure is handled by relief valves; however, to minimize the amount of steam/water so released, the steam generator contents are also dumped.

The steam generator interspace rupture discs are set at 825 psig (nominal). The burst pressure range (±2%) is 809 psig to 842 (Specification 2.2, Table 2.2-1). The relief valve is sized to allow a 370 psi pressure drop in a safety valve inlet line when the valve is relieving at nameplate capacity of 126,000 lb/hr superheated steam at 1000 degree F. This prevents the penetration pressure from exceeding the reference pressure of 845 psig.

Reheat Header Activity - High (Loop 1/Loop 2)

High reheat header activity is an indication of a reheater tube rupture resulting in leakage of reactor helium into the steam system. The trip setpoint assures detection of major reheat tube rupture and an on scale reading, with up to design value circulating activity for post accident monitoring. Detection of smaller size leaks or leaks with low circulating coolant activity can be detected and alarmed by the backup reheat condensate monitors and/or the air ejector monitor.

Low Superheat Header Temperature (Loop 1/Loop 2 and Differential)

Low superheat header temperature in a loop is indicative either of a feedwater valve or controller failure yielding an excessive loop feedwater flow or a deficiency of helium flow, and a loop shutdown is in order. The required coincident high differential temperature between loops functions to prevent the loop trip from occuring during normal operation at low main steam temperatures such as in a normal plant shutdown.

c) Circulator Shutdown Inputs

All circulator shutdown inputs (except circulator speed high on water turbines) are equipment protection items which are tied to two loop trouble through the loop shutdown system. These items are included in Table 3.3.1-3 because a malfunction could prevent a scram due to loss of the two loop trouble scram input. Circulator speed high on water turbines is included to assure continued core cooling capability on loss of steam drive.

DRAFT

APR 1 1985

Circulator Speed - Low

Too low a circulator speed causes a mismatch between thermal power input and heat removal (feedwater flow) in a steam generator, which may result in flooding the superheater section. The circulator trip causes an automatic adjustment, as required, in the turbine governor setting, feedwater flow, and remaining circulator speed to maintain stable steam pressure and temperature conditions.

Loop 1/Loop 2 Fixed Feedwater Flow - Low

The Fixed Feedwater Flow - Low is an equipment protection feature designed to protect the steam generator from overheating for complete loss of feedwater flow.

Loop 1/Loop 2 Programmed Feedwater Flow - Low

A Programmed Feedwater Flow - Low prevents prolonged operation in the region of speed versus flow which may cause excessive superheat steam temperatures.

Loss of Circulator Bearing Water

In order to prevent circulator damage upon loss of normal and backup bearing water supplies, a gas pressuized water accumulator is fired when water pressure falls below the trip setpoint value. The trip setpoint value is selected so that adequate water pressure is available during circulator coastdown, which lasts for about 30 seconds, to maintain clearances within the circulator bearings of at least 0.001 in. Tests and analyses have shown that a trip at 450 psid provides substantial clearance margin above 0.001 in. when the circulators are operating at normal speeds.

Circulator Penetration Trouble

Circulator penetration overpressure is indicative of a pipe rupture within the penetration. A circulator trip is appropriate for such an accident and the helium pressurizing line to the penetration is closed to prevent moisture backflow to the purified helium system. The overpressure is handled by the penetration relief valves. The penetration interspace rupture discs are set at 825 psig (nominal). The burst pressure range ($\pm$ 2%) is 809 psig to 842 psig (Specification 2.2 Table 2.2.1-1). The relief valve is sized to allow a 40 psi pressure drop in the safety valve inlet line when the valve is relieving at nameplate capacity (170 gpm).

Circulator Drain - Malfunction

This trip is provided to prevent steam from entering the bearing of an operating circulator. A differential pressure controller is utilized to maintain the bearing water main drain pressure above the steam turbine exhaust pressure. When the pressure differential drops, the steam water drain control valves are opened to prevent steam from entering the bearings. If the above controls do not work, three PPS differential pressure switches for each circulator, set at $\geq$ 8 psid, will initiate an automatic shutdown of the circulator.

DRAFT

APR 1 1985

Circulator Speed - High (Steam)

The speed sensing system response and trip setting are chosen so that under the maximum overspeed situation possible (loss of restraining torque) the circulator will remain within design capabilities.

Circulator Trip - Manual (Steam/Water)

A manual trip of each circulator for both steam and water turbine drives is available so that in an emergency an operator can trip a circulator when necessary.

Circulator Seal - Malfuncton (Low/High)

A high reverse differential of -6" H20 would be reasonable evidence that bearing water is leaking into the primary coolant system. An increasing differential pressure of +75.5" H20 would be reasonable evidence that primary coolant is leaking into the bearing water and thus into the closed circulator service system. In both cases a circulator trip with brake and seals set is appropriate.

Circulator Speed - High (Water)

The trip setpoint has been established above normal operating speed. Equipment testing ensures that this trip setpoint will prevent failure due to fatigue cracking.

d) Rod Withdrawal Prohibit Inputs

The termination of control rod withdrawal to prevent further reactivity addition will occur with the following conditions:

Start-up Channel - Low Count Rate

Start-up Channel-Low Count Rate is provided to prevent control rod withdrawal and reactor startup without adequate neutron flux indication. The trip level is selected to be above the background noise level.

Linear Channel -- 5% RWP

Linear Channel (5% Power) directs the operator's attention to either a downscale failure of a power range channel or improper positioning of the Interlock Sequence Switch.

Linear Channel - 30% RWP

Linear Channel (30% Power) is provided to prevent control rod withdrawal if reactor power exceeds the Interlock Sequence Switch limit for the "Low Power" position.

Start-up Channel/Wide Range Channel-Rate of Change - High

High Rate of change of neutron flux on the wide range channels ($\leq$2 dpm) initiates an RWP.

DRAFT

APR 1 1985

Linear Channel - High Power RWP

High neutron flux level from the power range channels initiates an RWP.

e) 480 VAC ESSENTIAL BUS UNDERVOLTAGE PROTECTION

Plant Electrical System - Loss

Each 480 VAC essential bus has three undervoltage relays set at ≥278 volts (58% of 480 volt nominal), arranged in two-out-of-three logic per bus and connected to a 30 ± 1.5 second time delay. Should power not be restored on two of the three 480 VAC essential buses before the 30 second time delay, a reactor scram will be initiated.

Degraded Voltage and Loss of Voltage

Protection under degraded conditions is provided for the essential power system by the use of undervoltage relays to produce appropriate action corresponding to the degree of voltage degradation.

Each 480 VAC essential bus has three undervoltage relays set at 416 ± 20 volts (86.7% of 480 volt nominal) arranged in two-out-of-three logic, one relay per phase. These relays are individually alarmed, and connected to 120 ± 5 second timers. In the event of a degraded power situation where two-of-the-three relays trip and remain tripped for the 120 second time period, the main power circuit breaker for that bus is opened. Timing out of the 120 second time delay which causes the affected 480 VAC essential bus to be deenergized, or voltage dropping below 372 ± 11 volts (77.5% of 480 volt nominal) in less than 120 seconds, will actuate inverse time delay undervoltage relays arranged in two-out-of-three logic per bus. Actuation of these inverse time delay relays will attempt to restore power to the affected bus by Automatic Throwover (ATO) to its neighboring 480 VAC essential bus. Interlocks are provided to prevent connecting more than two 480 VAC essential buses together. On a loss of 480 VAC bus voltage caused by the loss of all outside power, a second set of inverse time delay undervoltage relays set at 328 ± 10 volts (68.3% of 480 volts nominal) and arranged in two-out-of-three logic per bus will be tripped. Loss of voltage on two-out-of-three 480 VAC essential buses or tripping of two-out-of-three main power circuit breakers caused by the degraded voltage 120 second time delay will; 1) trip all three main power circuit breakers, 2) start the diesel generators, 3) load shed on all three buses, 4) close both diesel generator breakers and sequence the loads onto the diesel generators. The tie is established or re-established to the 480 VAC essential bus which is first energized by the diesel generator. Essential Bus 2 is interlocked so it can be connected to only one of the other two 480 VAC essential buses.

Time Allowed for Actions

The time allowed for specific ACTIONS have been selected to incorporate the intent of the standard technical specifications, while taking into account the unique design features of the FSV High Temperature Gas Cooled Reactor.

DRAFT
APR 1 1985

Time has been allowed immediately after discovery of an inoperable channel to permit repair without tripping the channel. Placing a channel in the tripped condition has been avoided as much as possible to reduce the possibility of unnecessary transients. The reduction of trips in this fashion while other protection functions are available will assure plant safety while minimizing challenges to plant safety systems.

Maintenance of inoperable channel(s) is to begin within two hours of discovery of the inoperable condition and surveillances are to be performed within 4 hours. These times are selected to permit calling out personnel and to permit set up of equipment. A minimum of 24 hours has been allowed to permit repair of a channel and/or shutdown of the reactor. In this manner an orderly shutdown of the reactor over a 24 hour period is available if it is determined that a channel cannot be repaired within the time permitted.

The specified surveillance check and test minimum frequencies are based on established industry practice and operating experience at conventional and nuclear power plants. The testing is in accordance with the IEEE Criteria (IEEE No. 279-1968) for Nuclear Power Plant Protection Systems, and in accordance with accepted industry standards.

Calibration frequency of the instrument channels listed in Tables 4.3.1-1, 4.3.1-2, 4.3.1-3, 4.3.1-4, and 4.3.1-5 are divided into three categories: passive type indicating devices that can be compared with like units on a continuous basis; semiconductor devices and detectors that may drift or lose sensitivity; and on-off sensors which must be tripped by an external source to determine their setpoint. Drift tests on transducers similar to the reactor pressure transducers (FSAR Section 7.3.3.2) indicate insignificant long term drift. Therefore a once per refueling cycle calibration was selected for passive devices (thermocouples, pressure transducers, etc.). Devices incorporating semiconductors, particularly amplifiers, will be also calibrated on a once per refueling cycle basis, and any drift in response or bistable setpoint will be discovered from the test program. Drift of electronic apparatus is not the only consideration in determining a calibration frequency; for example, the change in power distribution and loss of detector chamber sensitivity require that the nuclear power range system be calibrated every month. On-off sensors are calibrated and tested on a once per refueling cycle basis.

DRAFT

APR 1 1985

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

ANALYTICAL MOISTURE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.2.1 Primary coolant shall be sampled for moisture:

a. Upon entry and operation in STARTUP from SHUTDOWN, by two OPERABLE moisture monitors, normally from the Analytical System or alternately by use of the PPS dewpoint monitors, placed in the "Indicate" Mode*.

b. Upon entry and operation in STARTUP from LOW POWER, by one OPERABLE moisture monitor, normally from the Analytical System or alternately by use of the PPS dewpoint monitors placed in the "Indicate" Mode*.

APPLICABLILITY: STARTUP

ACTION:

a. Upon entry and operation in STARTUP from SHUTDOWN:

1. With only one OPERABLE moisture monitor, restore a second monitor to OPERABLE status or be in SHUTDOWN, LOW POWER* or POWER OPERATION* within 12 hours.

2. With no moisture monitors OPERABLE:

a.) Restore one monitor to OPERABLE status or be in SHUTDOWN, LOW POWER* OR POWER OPERATION* within 4 hours, and

DRAFT

APR 1 1985

b.) Restore a second monitor to OPERABLE status or be in SHUTDOWN, LOW POWER* or POWER OPERATION* within 12 hours of the first monitor being made OPERABLE.

b. Upon entry and operation in STARTUP from LOW POWER, with no moisture monitors OPERABLE, restore one monitor to OPERABLE status* or be in SHUTDOWN or LOW POWER within 4 hours.

* The use of the PPS dewpoint moisture monitors is contingent upon their availability for use in the "Indicate" Mode and the OPERABILITY requirements of Specification 3.3.1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 The analytical system primary coolant moisture instrumentation shall be demonstrated OPERABLE by:

a. Performance of CHANNEL CHECK less than 48 hours prior to entering STARTUP,

b. Performance of CHANNEL CHECK at least once per 48 hours during STARTUP,

c. Performance of CHANNEL CALIBRATION at least once per REFUELING CYCLE.

DRAFT

APR 1 1985

Basis for Specification LCO 3.3.2.1 / SR 4.3.2.1

While starting up or shutting down the plant, primary coolant moisture monitors are required below 5% reactor power for administration of Specification 3.4.3. One moisture monitor is sufficient to detect primary coolant moisture content on a continual basis.

Two analytical system moisture monitors (ME-9306, ME-9307) will normally be in service sampling primary coolant. These analytical moisture monitors do not provide any automatic protective action. Alternate moisture monitors can also be placed in service sampling primary coolant, such as through re-alignment of a moisture monitor in the analytical system or utilization of OPERABLE (as defined in Specification 3.3.1) plant protective system dewpoint moisture monitors placed in the "indicate" mode. Operator action is required to take corrective action in the event of high moisture levels in the primary coolant while the plant is in STARTUP.

Operator action to shut down the reactor in the event of high moisture levels in the primary coolant system at reactor power levels of 5% or less is acceptable. As indicated by Figure 4.2 in Document GA-A13677, Test and Evaluation of the Fort St. Vrain Dew Point Moisture Monitors System, one of the limiting parameters for determining required response times to shut the reactor down in the event of high primary coolant moisture is graphite oxidation. The allowable weight loss of the hottest fuel element in the core is 1%.

At operating temperatures experienced at 5% reactor power, response time to scram the reactor to limit oxidation to 1% by weight is approximately 1 hour and 50 minutes, well within the capabilities of an operator.

The 4 hour ACTION times are based on the time required to perform corrective measures as necessary (gain access, re-establish sample flow, allow instruments to stabilize etc.) to initiate use of an alternate moisture monitor or perform limited maintenance on the inoperable monitor(s).

The surveillance interval specified for checking OPERABILITY and for calibration of this instrumentation will assure the proper operation of these detectors.

DRAFT

APR 1 1985

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.2 The radiation monitoring instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.2-1

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3.2-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3.2-1.

c. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.2 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3.2-1.

DRAFT

APR 1 198

TABLE 3.3.2-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Control Room: (RT-93250-5)	1	ALL MODES	less than or equal to 1.5 mR/hr	1 E-1 to 1 E +4 mR/hr	1
2. PROCESS MONITORS					
a. Steam/Water Dump Tank Monitors					
Radiation Shine (RT-93250-12, RT-93251-12)	1	P,L	less than or equal to 3.0 mR/hr	1 E-1 to 1 E+3 mR/hr	2
b. PCRV Relief Valve Piping Monitor					
Radiation Shine (RT-93252-12)	1	P,L,S/U	less than or equal to 3.0 mR/hr	1 E-1 to 1 E+3 mR/hr	2
3. ACCIDENT MONITORING					
a. Reactor Building Accident Monitoring					
Refueling Floor-East Wall (RT-93250-14)	1	P,L,S/U	less than or equal to 3.0 R/hr	1 E-1 to 1 E+4 R/hr	2
b. Reactor Plant Exhaust Filter Monitoring (RT-93251-1)	1	P,L,S/U	Less than or equal to 2.0 R/hr	1 E-1 to 1 E+4 R/hr	2

DRAFT
APR 1 1985

TABLE 3.3.2-1 (Continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, place a portable monitor equipped with an alarm in the area within 6 hours and notify potentially affected personnel.

ACTION 2 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel(s) to OPERABLE status within 7 days, or be in SHUTDOWN within the subsequent 36 hours.

DRAFT

APR 1 1985

TABLE 4.3.2-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a) Control Room (RT-93250-5)	D	R	M	ALL MODES
2. PROCESS MONITORS				
a) Steam/Water Dump Tank Monitors (RT--93250-12, RT-93251-12)	D	R	M	P,L
b) PCRV Relief Valve Piping Monitor (RT-93252-12)	D	R	M	P,L,S/U
3. ACCIDENT MONITORING				
a) Reactor Building Accident Monitoring (RT-93250-14)	D	R	M	P,L,S/U
b) Reactor Plant Exhaust Filter (RT-93251-1)	D	R	M	P,L.S/U

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.3.2.2 / SR 4.3.2.2

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are measured in the areas served by the individual channels and 2) an alarm is initiated when the radiation level trip setpoint is exceeded. The ACTIONS are commensurate with industry practice for this type of equipment.

Process and area radiation monitoring systems that are required to protect the health and safety of the public are listed in FSAR Table 7.3-2 (FSAR 7.3.5.3). Additionally, NUREG 0737 Item II.F.1.3 requires accident-monitoring; containment high-range monitor. PSC installed a high-range reactor building radiation monitor (refueling floor-east wall) with an upper limit of 1 E+4 R/hr (RT-93250-14). The refueling floor-east wall monitor (RT-93250-14) and the reactor plant exhaust filter monitor (RT-93251-1) are used for Design Basis Accident Number 2, Rapid Depressurization of the Primary Coolant System. (Ref. FSAR Amendment 24 Attachment B).

The basis for the instrument groupings in Table 3.3.2-1 is as follows:

1. The Area Monitor given in Table 3.3.2-1 is the main control room area monitor which relates to control room habitability (the actual control room ventilation system recirculation control is accomplished by the Reactor Building Ventilation Exhaust Monitors (FSAR Section 7.3.5.2)).

2. Process Monitoring addressed elsewhere in the Technical Specifications, as indicated below, are not included in Table 3.3.2-1. The Process Monitoring given in FSAR Table 7.3-2 is addressed in the following manner*:

   A. Table 3.3.2-1 includes the Steam/Water Dump Tank Monitors and PCRV Relief Valve Piping Monitor, which are not addressed elsewhere in the Technical Specifications.

   B. Radioactive Gaseous Effluent Monitoring includes the following, for which requirements are given in Specification 8.1.1. This includes control room ventilation system recirculation control on high radiation (FSAR Section 7.3.5.2).

* Whi'ɔ Table 7.3-2 is titled Process and Area Radiation Monitors, they are herein functionally considered to all be Process Monitors).

DRAFT

APR 1 1985

1) Ventilation Exhaust Monitors - RT-7324-1, -2
RT-73437-1, -2
RT-4801-1, -2, -3
RT-7325-1, -2

2) Gas Waste Exhaust - RT-6314-1, -2

3) Secondary Coolant Air Ejector- RT-31193

C. Radioactive Liquid Effluent Monitoring includes the following, for which the requirements are given in Specification 8.1.2 and Specification 8.1.3.

1) Radioactive Liquid Waste Discharge - RT-6212, 6213

2) Gas Waste Compressor Cooling Water Activity - RT-46211, 46212

D. The Secondary Coolant Reheat Steam Monitors (RT-93250-10, -11; RT-93251-10, -11; and RT-93252-10, -11) are included as part of the P.P.S. loop shutdown (Specification 3.3.1).

E. The reheater/steam generator interspace process monitors (RT-2263 and RT-2264) have requirements as specified in Specification 3.6.1.5.

3. The Accident Monitoring included in Table 3.3.2-1 involves the high range reactor building radiation monitor (RT-93250-14) and the reactor plant exhaust filter monitor (RT-93251-1).

The surveillance interval specified for CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION are adequate to assure the proper operation of these detectors.

DRAFT
APR 1 1985

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.2.3 The seismic monitoring instrumentation shown in Table 3.3.2-2 shall be OPERABLE

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE seismic monitoring instruments less than the minimum INSTRUMENTS OPERABLE requirement for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.3.1 Each of the seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.2-2

4.3.2.3.2 Each of the seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and the calibration of the vertical seismic triggers shall be checked on-site within 5 days following the seismic event (out of tolerance vertical seismic triggers shall be defined inoperable). Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. Upon the actuation of a Seismic Trigger due to a seismic event, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days describing the magnitude, frequency, and resultant effect upon facility features important to safety.

DRAFT
APR 1 1985

TABLE 3.3.2-2

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		2
a. PCRV Support Ring	-1 to +1g	
b. Top of PCRV	-1 to +1g	
c. Visitors Center	-1 to +1g	
2. Vertical Seismic Triggers		2
a. PCRV Support Ring*	Set Point less than or equal to 0.015g	
b. Top of PCRV*	Set Point less than or equal to 0.015g	
c. Visitors Center	Set Point less than or equal to 0.015g	
3. Seismoscopes		2
a. PCRV Support Ring	N/A	
b. Top of PCRV	N/A	
c. Visitors Center	N/A	

*With control room alarm

DRAFT

APR 1 1985

TABLE 4.3.2-2

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelerographs			
a. PCRV Support Ring	M*	18 Mos.	SA
b. Top of PCRV	M*	18 Mos.	SA
c. Visitors Center	M*	18 Mos.	SA
2. Vertical Seismic Triggers			
a. PCRV Support Ring**	N/A	18 Mos.	SA
b. Top of PCRV**	N/A	18 Mos.	SA
c. Visitors Center	N/A	18 Mos.	SA
3. Seismoscopes			
a. PCRV Support Ring	M*	A	N/A
b. Top of PCRV	M*	A	N/A
c. Visitors Center	M*	A	N/A

*Except seismic trigger

**With control room alarms

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.3.2.3 / SR 4.3.2.3

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to determine the magnitude of a seismic event and to sound an alarm in the event that a disturbance greater than the setpoint is experienced. (The minimum instruments OPERABLE also reflects the need to send instruments off site for calibration.) This capability permits comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown or inspection is necessary pursuant to FSAR Section 7.3, and the plant emergency procedures.

The nominal setpoint for the vertical seismic triggers is 0.01g, the 0.015g figure in Table 3.3.2-2 reflects instrumentation calibration tolerances.

The intervals specified for testing and calibration of the Seismic Instrumentation are considered adequate to assure the instruments operate as intended.

The time-history accelerographs including seismic triggers are sent to the manufacturer for calibration on an 18 month cycle (in accordance with the manufacturer's recommendation). The calibration of the seismic triggers is additionally checked on site. This on-site calibration check will also be performed following seismic events which cause the instruments to be actuated. Instruments so determined to be out of calibration will be sent to the manufacturer for CHANNEL CALIBRATION.

The Special Report required in Specification 4.3.2.3.2 to be submitted to the commission following a seismic event does not include response spectrums data. Response spectrum data, when deemed necessary, will be obtained by off-site digitization of the film data and subsequent data reduction which requires several weeks.

DRAFT

APR 1 1985

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.4 The meteorological monitoring instrumentation channels shown in Table 3.3.2-3 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE meteorological monitoring channels less than the MINIMUM OPERABLE requirement for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.4 The meteorological monitoring instrumentation channels required above, shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.2-3.

DRAFT

APR 1 1985

TABLE 3.3.2-3
METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM OPERABLE
1. WIND SPEED	
a. Primary Tower - 10 meters	1*
b. Primary Tower - 58 meters	1
2. WIND DIRECTION	
a. Primary Tower - 10 meters	1*
b. Primary Tower - 58 meters	1
3. DELTA TEMPERATURE	
a. Primary Tower - 10/58 meters	1

* The backup (NOAA 10 meter) tower may be used as an acceptable alternate channel for these parameters.

DRAFT

APR 1 1985

TABLE 4.3.2-3

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Primary Tower - 10 meters	D	SA
b. NOAA 10 Meter Tower - 10 meters	D	SA
c. Primary Tower - 58 meters	D	SA
2. WIND DIRECTION		
a. Primary Tower - 10 meters	D	SA
b. NOAA 10 Meter Tower - 10 meters	D	SA
c. Primary Tower - 58 meters	D	SA
3. DELTA TEMPERATURE		
a. Primary Tower - 10/58 meters	D	SA

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.3.2.4 / SR 4.3.2.4

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental releases of radioactive materials to the atmosphere. This instrumentation provides the capability to evaluate the need for initiating proper protective measures to protect the health and safety of the public and, as outlined in PSC letter P-82400, reflects the guidance given in Proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants," September 1980. The primary data for accident release dose assessment calculations utilizes wind speed at 10 meters, wind direction at 10 meters and delta temperature (for atmospheric stability). The remaining measurements serve as backups. The backup (NOAA 10 meter) tower may be used to meet the minimum operable requirement channels requirement as shown on Table 3.3.2-3.

The intervals specified for testing and calibration of the meteorological instrumentation provide assurance of instrumentation OPERABILITY and are consistent with nuclear plant practice and recommendations given in Proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants," September, 1980.

DRAFT
APR 1 1985

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

FIRE DETECTION AND ALARM SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.2.5 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.2-4 shall be OPERABLE.

APPLICABILITY: At all times

ACTION: With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3.2-4:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour.

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days, or in lieu any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperablility and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.5.1 Each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.2.5.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above

DRAFT

APR 1 1985

required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.2.5.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

DRAFT
APR 1 1985

TABLE 3.3.2-4

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE HEAT	SMOKE
1. Control Room		4
2. Auxiliary Electric Room		4*
Return Air Duct		1
3. 480 V Switchgear Room		4*
4. Reactor Bldg "J" Wall		
Elevation 4756 to 4791		2
Elevation 4791 to 4829		2
Elevation 4829 to 4849		1
Elevation 4849 to 4881		1
5. Turbine Bldg "G" Wall		
Elevation 4791 to 4811		1
Elevation 4811 to 4829		1
6. Refueling Floor HVAC Intake		1
7. Reactor Building HVAC Return Air Ducts		6

*Detector(s) automatically actuate fire suppression system(s)

DRAFT
APR 1 1985

TABLE 3.3.2-4 (Continued)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE HEAT	SMOKE
8. Building 10		
Switchgear Room & Ground Level		4*
Ground Level Under Mezzanine		4*
Battery Room		1*
9. Radwaste Compacting & Storage Area		1
10. Hydraulic Valve Area (Level 6)		4
11. Hydraulic Power Unit (Level 1)		1
Oil Mist Detectors		1(Oil Mist Detector)
12. Turbine Plant MCC2 & MCC3		1
13. Service Water Pump Building		1
14. Circulating Water Makeup Pump Building		1
15. Reactor Plant Exhaust Filter		
Filter 1A	2*	
Filter 1B	2*	
Filter 1C	2*	
16. Diesel Generator Rooms		
Room 1A	1*	
Room 1B	1*	

*Detector(s) automatically actuate fire suppression system(s)

DRAFT

APR 1

BASIS FOR SPECIFICATION LCO 3.3.2.5 / SR 4.3.2.5

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

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

The smoke detection and alarm systems provide detection and alarm capability for the control room, auxiliary electric room, the 480 volt switch gear room, the congested cable areas located at the "G" and "J" walls, selected reactor building HVAC return air ducts and selected safety related equipment items/areas.

Smoke detection will automatically initiate operation of the Halon fire suppression system in the auxiliary electric room, the 480 volt switch gear room, or Building 10 safety related equipment areas.

Heat sensors associated with the Reactor Building Exhaust Filters and Diesel Generator Rooms will automatically initiate operation of water spray and carbon dioxide suppression systems respectively.

The minimum number of operable detectors for a given area depends upon the number of detectors in the area and typically is at least one half the total number in the area.

The surveillance interval specified for this instrumentation is consistent with industry practice for this type equipment and assures its proper operation in the event of a fire.

DRAFT
APR 1 1985

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

CHLORINE DETECTION AND ALARM SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.2.6 The chlorine detection and alarm system shall be OPERABLE to alert control room personnel in the event of an accidental chlorine release.

APPLICABILITY: At all times

ACTION:

a. With either the detection or alarm system inoperable, restore the inoperable system to OPERABLE status within 24 hours or within one hour initiate and maintain operation of the control room emergency ventilation system in the minimum makeup mode of operation.

b. The provisions of Specification 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.6 The chlorine detection and alarm system shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 92 days, and by a CHANNEL CALIBRATION at least once per 18 months.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.3.2.6 / SR 4.3.2.6

The OPERABILITY of the chlorine detection and alarm system ensures the capability to detect an accidental chlorine release and alert control room personnel so that protective actions may be initiated to maintain control room habitability.

The surveillance interval specified for this instrumentation assures proper operation in the event of an accidental chlorine release.

DRAFT

APR 1 1985

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

POWER-TO-FLOW RATIO RECORDING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.7 The POWER-TO-FLOW RATIO recording instrumentation shall be OPERABLE.

APPLICABILITY: POWER OPERATION and LOW POWER*

ACTION:

a. With the above required instrumentation inoperable:

1. Ensure within 24 hours the operability of a back-up means of recording data for computing the POWER-TO-FLOW RATIO, and

2. Restore the recording instrumentation to OPERABLE status within 7 days, or

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.7 The POWER-TO-FLOW RATIO recording instrumentation shall be demonstrated OPERABLE:

1. At least once per 24 hours by performance of a CHANNEL CHECK,

2. At least once per REFUELING cycle by performance of a CHANNEL CALIBRATION.

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* Above 15% RATED THERMAL POWER

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.3.2.7 / SR 4.3.2.7

The POWER-TO-FLOW RATIO instrumentation gives an indication of the balance between the heat generation and removal within the primary coolant system.

The OPERABILITY of the recording instrumentation ensures that sufficient capability is available to determine the magnitude of a transient event affecting the POWER-TO-FLOW RATIO. This capability permits a computer calculation or manual computation of the integrated time during the measured POWER-TO-FLOW RATIO transient and comparison with the allowable limits in Specification 2.1.1.

The Data Acquisition System can provide backup for the strip-chart recorder when the strip-chart recorder is out of service.

A verification of the POWER-TO-FLOW recording on a daily basis during LOW POWER and POWER OPERATION is adequate to assure the instrument is recording properly. In addition, any change in reactor power level, no matter how small, should produce a change in the POWER-TO-FLOW RATIO indication. Calibration of the instrumentation on a once-per-refueling cycle basis is acceptable by industry standards for this type of equipment.

"APPLICABILITY" is limited to power levels above 15% of RATED THERMAL POWER, in that the POWER-TO-FLOW RATIO recorder is only required in the range of 15% to 100% power to assure compliance with Specification 2.1.1. Specification 3.2.4 "Core Inlet Orifice Valves /Minimum Helium Flow", applies to power levels below 15%, where core temperatures are lower, and the P/F recorder is not required.

DRAFT

APR 1 1985

INSTRUMENTATION

3/4.3.3 THREE ROOM CONTROL COMPLEX TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3 The temperature of each area shown in Table 3.3.3-1 shall be maintained within the limits indicated.

APPLICABILITY: At all times

ACTION: With one or more areas exceeding the temperature limit(s) shown in Table 3.3.3-1:

a. Within one hour, either restore the normal HVAC or initiate establishment of temporary cooling to the affected area.

b. For more than 8 hours, in lieu of any Licensee Event Report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

c. By more than 30 degrees F, within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable. In lieu of any Licensee Event Report required by Specification 6.9.1, provide a Special Report as required by above ACTION b.

SURVEILLANCE REQUIREMENTS

4.3.3 The temperature in each of the areas shown in Table 3.3.3-1 shall be determined to be within its limit at least once per 24 hours.

DRAFT

APR 1 1985

TABLE 3.3.3-1

THREE ROOM CONTROL COMPLEX

TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (degrees F)
1. Control Room	115
2. Auxiliary Electric Equipment Room	115
3. 480 Volt Switchgear Room	115

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.3.3 / SR 4.3.3

The limiting temperatures in the THREE ROOM CONTROL COMPLEX are established to assure no over temperature condition which might cause damage to essential instrumentation and control equipment. Satisfactory operation of safety related control and electrical equipment located in the three room control complex for temperatures up to 120 degrees F is discussed in FSAR Section 7.4.1.

Exposure to excessive temperatures may degrade equipment and may cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of 5 degrees F.

The surveillance interval specified assures adequate temperatures are maintained in the designated plant areas.

DRAFT

APR 1 1985

PRIMARY COOLANT

3/4.4.1 PRIMARY COOLANT ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.1 The primary coolant gaseous and plateout activity levels shall be limited to:

a. 2.40 curies-mev/lb, which is the product of primary coolant noble gas beta plus gamma activity times E-Bar, corrected to 15 minutes after sampling,

b. 24 curies of thyroid DOSE EQUIVALENT I-131 circulating in the primary coolant,

c. 5000 curies per loop of thyroid DOSE EQUIVALENT I-131 plateout within the primary circuit, and

d. 140 curies per loop of SR-90 plateout within the primary circuit.

APPLICABILITY: POWER OPERATION LOW POWER and STARTUP

ACTION:

a. With the primary coolant circulating gaseous activity exceeding above limits a. or b., or with the plateout iodine inventory exceeding limit c. either be in STARTUP within 12 hours and be in SHUTDOWN within the following 12 hours, or initiate the following:

1. Reduce thermal power by at least 10% of RATED THERMAL POWER within 12 hours and

2. At least once per 24 hours, perform sampling and analysis in accordance with Specification 4.4.1, and

3. If, after a maximum of 48 hours at the reduced power level, limit a., b., or c. continues to be exceeded, either:

a) Repeat ACTION a.1. within 12 hours, followed by ACTION a.2. and a.3. until a power level is reached at which limit a., b. and c. are met, or

b) Be in STARTUP within the subsequent 12 hours, and be in SHUTDOWN within the following 12 hours.

c) With the primary coolant Sr-90 plateout activity exceeding limit d. above, be in STARTUP within 12 hours and be in SHUTDOWN within the following 12 hours.

d) With the primary coolant gross activity monitor or associated recorder inoperable, reactor operation may continue provided a primary coolant sample is collected and analyzed at least once per 24 hours per Specification 4.4.1.

e) The provisions of Specification 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1 The primary coolant gaseous and plateout activity shall be determined to be within the above limits as follows:

a. At least once per 24 hours by use of the gross gamma activity monitor,

b. At least once per 14 days by taking and analyzing a grab sample of primary coolant. This grab sample analysis shall be used to verify the sensitivity of the gross gamma activity monitor, and shall determine the following:

1. E-Bar (E) (See NOTE 1)

2. Curies - mev/lb

3. Circulating and plateout curies of DOSE EQUIVALENT I-131,

4. Total plateout SR-90 inventory.

c. With the gross gamma activity monitor inoperable, at least once per 24 hours a grab sample of primary coolant shall be taken and analyzed for the parameters listed in b. above.

d. If the activity level reaches 25% of a limit above, at least once per 24 hours a grab sample of primary coolant shall be taken and analyzed for the parameters listed in Specification 4.4.1.b above. Normal sample frequency (i.e., at least once per 14 days) may be resumed when the activity level is reduced to below 25% of the limits, or when the activity level reaches a new equilibrium value, as defined by three consecutive daily samples whose results agree within 10% of the average of the three samples.

e. One plateout probe shall be removed for evaluation coincident with the second, fourth, and sixth refueling*, and at intervals not to exceed five refueling cycles* thereafter. If, during the third or fifth refueling cycle, or any refueling cycle following the sixth refueling, 25% of above limits Specification 3.4.1.c or 3.4.1.d is reached, or the primary coolant activity is greater than 25% of Design, activity FSAR table 3.7.-1 the plateout probe shall be removed at the end of that refueling cycle. The probes shall be analyzed for Sr-90 inventory in the primary circuit. The probes removed shall also be analyzed for I-131. The results shall be used to verify the plateout activity of SR-90 and I-131 in the primary circuit, as determined by Specification 4.4.1.b.3 and Specification 4.4.1.b.4 above.

NOTE 1: Calculations required to determine E-Bar will consist of the following:

1. Quantitative measurement in units of Ci/lb of radionuclides making up at least 95% of the noble gas beta plus gamma decay energy in the primary coolant corrected to 15 minutes after sampling.

2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above, by applying known decay energies and schemes.

3. A calculation of E-Bar by appropriate weighting of each nuclide's beta and gamma energy with its concentration as determined in (1) above.

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* A "refueling" follows the "refueling cycle" with the same number - i.e., the sixth refueling follows the sixth refueling cycle.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATIONS LCO 3.4.1 / SR 4.4.1

Basis for Noble Gas Beta plus Gamma Activity Limit

The whole body dose is a direct function of the gross gamma activity in the primary coolant. The whole body skin dose is a direct function of the gas beta activity in the primary coolant.

The Specification 3.4.1.a limit on the primary coolant noble gas beta plus gamma concentration is based on the Maximum Credible Accident or MCA (FSAR Section 14.8) wherein the entire "Design" primary coolant circulating gaseous radioactive inventory is carried out of the PCRV and is released to the atmosphere through the plant vent system.

Correcting the primary coolant beta plus gamma activity to 15 minutes after sampling would conservatively indicate the activity that would reach the Exclusion Area Boundary (EAB), following the postulated accident, taking into account the decay of short half-life isotopes during atmospheric transport to the EAB.

The U.S. Atomic Energy Commission Staff (see Table 4.1 of Ref. 1) used a number of conservative assumptions to calculate the MCA doses at the EAB. These conservatisms included a short-term atmospheric dilution factor of 2.6 E-3 sec/m3 resulting from an assumed downdraft of the exhaust plume at a wind speed of only 0.3 m/sec during Pasquill atmospheric condition F. This produced a whole body dose for the MCA of 8.6 rem at the EAB, which is well below the 10CFR Part 100 guidelines.

Basis for Iodine and Strontium Activity Limits

The SR-90 and thyroid DOSE EQUIVALENT I-131 limits are based on the Design Basis Accident No 2 (PCRV rapid depressurization-FSAR Section 14.11) wherein the entire primary coolant circulating inventory, and fractions of the plateout iodines and strontium are carried out of the PCRV and out of the reactor building through the louvers.

The U.S. Atomic Energy Commission Staff (see Table 4.2 of Ref. 1) used a number of conservative assumptions to calculate the accident consequences. However, these maximum equivalents result in calculated EAB doses which are well below 10 CFR 100 limits. The maximum equivalent activity levels (i.e., Sr and I-131 limits) determined by the Commission Staff from the Design Basis Accident No. 2 (DBA-2) are summarized in the following table:

DRAFT

APR 1 1985

LCO Activity Levels Determined by
The Depressurization Accident (DBA-2)

Dose Category	Nuclide Equivalent	LCO Limit Max. Equiv. Curies Plated Out	% Released to Environ.	Resulting EAB Dose(rem)
THYROID:				
Plateout	I-131	5000/loop	6%	139
Circulating	I-131	24(Not Plated Out)	100%	11
Total Thyroid Dose				150
BONE	SR-90	140/loop	5%	75

The activity levels shown are based on the resulting doses at the EAB, assuming a dilution factor of 8.4 E-4 sec/m3 and effectivities of 1,480 rem per milli-Curie of I-131 inhaled, and 36,700 rem per milli-Curie of Sr-90 inhaled. These effectivity values are based on information in ICRP II. The newer data, especially for Sr-90 given in the more recent ICRP VI, were ignored.

Should information become available which leads to a change in the given dilution factors (e.g. Ref. 2), or should the data given in ICRP VI become acceptable, the allowable activity concentrations (LCO limits) may be changed accordingly.

Action Statement Bases

ACTION a. statement permits graduated (10% or more) reduction in power level in the event circulating activity exceeds limits followed by up to 48 hours for sampling. The 48 hour sampling delay allows time for the activity level to decrease (due to decay, cleanup, and lower fuel temperatures) and reach a new equilibrium level.

ACTION b. statement does not permit critical reactor operation with the SR-90 plateout activity greater than 140 Ci/loop. A shutdown is appropriate.

ACTION c. statement sampling and analysis interval of 24 hours is reasonable and prudent to ensure no significant changes in fuel performance characteristics will occur undetected, since the HTGR is not subject to sudden large increases in primary coolant activity level during normal operation.

Basis For SR 4.4.1 (b), (c) and (d)-Sampling and Analysis

Under normal operating conditions the gross gamma activity of the primary coolant is measured and indicated on a continuous basis. The 14 day sampling interval provides an adequate check on the sensitivity of this monitoring equipment. If the gross activity monitor becomes inoperable, within 24 hours a sample will be taken and analyzed, and this will be continued on a 24 hour interval basis until the monitor becomes operable. The 24 hour sampling and analysis interval is reasonable and prudent to ensure no significant changes in fuel performance characteristics will occur undetected, since the HTGR is not subject to sudden large changes in primary coolant activity level during normal operation.

The noble gas inventories are calculated from grab samples and the readings of the gross gamma monitor. It has been demonstrated by theoretical investigations and experiments that the steady state release rate of noble gas fission products from failed fuel particles is approximately proportional to the square root of the fission product half-life. Further information is given in Section 3.7 of the FSAR. The inventory of any non-measured noble gas nuclide (and iodine nuclides necessary to calculate thyroid DOSE EQUIVALENT I-131 inventories) and Sr-90 inventories is calculated by assuming that the release rate is predictable from the release rate curves established using the xenon and krypton noble gas nuclides.

Basis For SR 4.4.1 (e)-Plateout Probe Surveillance

The plateout probes are located in penetrations extending into steam generator shrouds and then into the gas stream of each coolant loop. One sample is accumulated by continuously bypassing a small portion of the core outlet coolant stream through diffusion tubes and sorption beds located in the probe body. Another sample can be accumulated by continuously bypassing a portion of the circulator outlet coolant stream through the probe. The core outlet sample can be used to determine the concentrations of fission products in the coolant stream entering the steam generator; the circulator outlet sample provides information about the amount of cleanup in each pass around the circuit. The analysis for I-131 is made to determine the degree of conservatism of the assumptions made regarding the circulating and plateout iodine in the primary coolant circuit.

The interval for probe removal and analysis subsequent to the sixth refueling cycle may be adjusted based upon the analysis of prior results.

DRAFT

APR 1 1985

The Sr-90 inventory is determined by an analysis of the plateout probes. In the interim between probe removals, the Sr-90 inventory may be tentatively estimated from

$$A_{Sr\text{-}90}(t) = A_{Sr\text{-}90}(0)e^{-\text{lambda } t} + \text{integral}_{0}^{t} A_{Kr\text{-}90}(\text{tau})e^{-\text{lambda } (t-\text{tau})} \text{ lambda d tau}$$

where A Sr-90 (0) is the total Sr-90 inventory in the loop, as determined by the most recent plateout probe analyses, t is the elapsed time since this determination. lambda is the decay constant for Sr-90, and A Kr-90 (tau) is the time dependent Kr-90 activity in the coolant stream based on release-rate/birth-rate curves obtained from the analyses of the bi-weekly grab samples. Note that, if the Kr-90 activity is constant (or bounded, or can be averaged), the estimated Sr-90 inventory would be given by

$$A_{Sr\text{-}90}(t) = A_{Sr\text{-}90}(0)e^{-\text{lambda } t} + \text{A-bar}_{Kr\text{-}90}\,(1 - e^{-\text{lambda } t}).$$

This method of estimating the Sr-90 inventory in the interim between probe removals is based upon the consideration that the source of Sr-90 is anticipated to be predominantly from Kr-90. However, the Sr-90 inventory will be periodically updated by the plateout probe analyses to give the total measured Sr-90 plateout, regardless of origin.

REFERENCES

1. Safety Evaluation by the Division of Reactor Licensing, U. S. Atomic Energy Commission, in the Matter of Public Service Company of Colorado, Fort St. Vrain Nuclear Generating Station, Docket No. 50-267, issued: January 20, 1972

2. Regulatory Guide 1.145 - "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, reissued February, 1983.

DRAFT

APR 1 1985

PRIMARY COOLANT

3/4.4.2 LOOP IMPURITY LEVELS - HIGH TEMPERATURES

LIMITING CONDITION FOR OPERATION

3.4.2 The chemical impurity concentrations in the primary coolant shall not exceed 10 ppm (by volume) for the sum of H_2O, CO, and CO_2.

APPLICABILITY: Whenever CORE AVERAGE OUTLET TEMPERATURE is 1200 degrees F or greater.

ACTION: With chemical impurity concentrations (sum of H_2O, CO, and CO_2) exceeding:

a. 10 ppm but less than or equal to 100 ppm - reduce chemical impurity concentrations to 10 ppm or less within 10 days from the time the limit is exceeded or decrease CORE AVERAGE OUTLET TEMPERATURE to less than 1200 degrees F within the next 6 hours and comply with Specification 3.4.3.

b. 100 ppm but less than or equal to 1000 ppm - reduce chemical impurity concentration to below 100 ppm within 24 hours from the time the limit is exceeded or be below 1200 degrees F CORE AVERAGE OUTLET TEMPERATURE within the next 6 hours and comply with Specification 3.4.3.

c. 1000 ppm - reduce CORE AVERAGE OUTLET TEMPERATURE to below 1200 degrees F within 1 hour from the time the limit is exceeded and comply with Specification 3.4.3.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.4.2.1 CHEMICAL IMPURITIES

When the reactor is critical, the primary coolant shall be analyzed for H_2O, CO, and CO_2:

a. At least once per 24 hours when the chemical impurity levels exceed 50 percent of the limits, or

b. At least once per 7 days at all other times.

4.4.2.2 PGX GRAPHITE SURVEILLANCE

The PGX graphite surveillance specimens (16 per reflector element) installed in reflector elements as indicated in Table 4.4.2.2-1 shall be removed at refueling intervals shown in Table 4.4.2.2-1 unless the progressive examination of the specimens dictate otherwise.

Upon removal, these specimens shall be subjected to examination, and compared with laboratory control specimens in evaluating oxidation rates, oxidation profiles, and general dimensional characteristics.

The results of these tests and examinations shall be utilized to assess the condition of the PGX core support blocks in the reactor and shall also be utilized to modify, as necessary, the planned removal of subsequent PGX surveillance specimens.

The results of these examinations shall be submitted to the NRC staff for review.

4.4.2.3 CORE SUPPORT BLOCK SURVEILLANCE

The top surface of the core support block for fuel regions fitted with PGX graphite specimens (Regions 22, 24, 25, 27 & 30) shall be visually examined by remote TV for indication of cracks, in particular in areas where analysis shows the highest tensile stresses exist, at the refueling shutdown when the PGX graphite specimens are removed from the core in accordance with Technical Specification SR 4.4.2.2.

DRAFT

APR 1 1985

PGX GRAPHITE SURVEILLANCE

TABLE 4.4.2.2-1

REFLECTOR ELEMENT ASSEMBLY WITHDRAWAL SCHEDULE

Fuel Region	Column	Withdrawal at Refueling Number*
25	7	2
30	3	4
24	7	6
22	6	9
27	2	17

*Schedule would be adjusted to remove transition element assemblies at a faster rate should specimens at any withdrawal interval show a burnoff significantly greater than predicted.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.4.2 / SR 4.4.2.

For plant operation in the range of about 25% to 100% of RATED THERMAL POWER, maximum impurity levels have been established to restrict carbon transport from the reactor core to cooler portions of the primary coolant system to about 330 lb/yr.

Limiting the quantity of carbon transported from the reactor core insures the integrity of the fuel elements, insures the integrity of the core support structure, and limits the effect on the steam generator heat transfer properties. The carbon corrosion will be fairly uniformly distributed throughout the outlet third of the core, resulting in a rate of weight loss from this portion of the core of about 0.3% per year (see FSAR Section 9.4.2).

PGX graphite specimens have been placed in modified coolant channels in five (5) transition reflector elements in the hottest columns of regions 22, 24, 25, 27, and 30. The surveillance test specimens will be subjected to the primary coolant conditions, as well as other reactor parameters that are normally seen by the PGX core support blocks. Examination and tests of the surveillance test specimens are regular intervals can readily be utilized to assess oxidation rates, oxidation profiles, as well as general degradation of the PGX core support blocks to adequately predict the structural integrity of the core support blocks over the operating life of the reactor.

Visual examination of the core support blocks in those regions chosen for insertion of PGX graphite speciments will provide additional assurance that integrity of the core support blocks does not degrade due to plant operating conditions, since those regions were selected because of their higher potential for PGX graphite burnoff. Analysis shows that the highest tensile stresses occur on the top surface of the core support blocks, at the keyways, and at the web between reactor coolant channels.

DRAFT

APR 1 1985

PRIMARY COOLANT

3/4.4.3 LOOP IMPURITY LEVELS - LOW TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.4.3 Primary coolant chemical impurity concentrations shall not exceed the following:

a.

IMPURITY	LIMIT
H2O	The dew point limits shown in Figure 3.4.3-1 as Acceptable or Limited Acceptable.
CO2	1,000 ppm (by volume)
CO	15,000 ppm (by volume)

b. In addition, during reactor startups and shutdowns with CORE AVERAGE OUTLET TEMPERATURE between 725 degrees F and 1200 degrees F, the cumulative time during which the dew point exceeds -20 degrees F shall not exceed a total of 90 days during any one REFUELING CYCLE. The operating conditions under which this limitation applies are shown as the "Limited Acceptable" region on Figure 3.4.3-1.

APPLICABILITY: Whenever CORE AVERAGE OUTLET TEMPERATURE is less than 1200 degrees F.

DRAFT

APR 1 1985

ACTION:

a. With primary coolant impurity concentrations exceeding the specified limits, reduce the concentrations to within "Acceptable" or "Limited Acceptable" values per Figure 3.4.3-1 within 1 hour or reduce CORE AVERAGE OUTLET TEMPERATURE below 725 degrees F within the next 6 hours.

b. With primary coolant moisture dew point and CORE AVERAGE OUTLET TEMPERATURE in the "Limited Acceptable" region of Figure 3.4.3-1, limit the cumulative time under these conditions to a total of 90 days during any one REFUELING CYCLE.

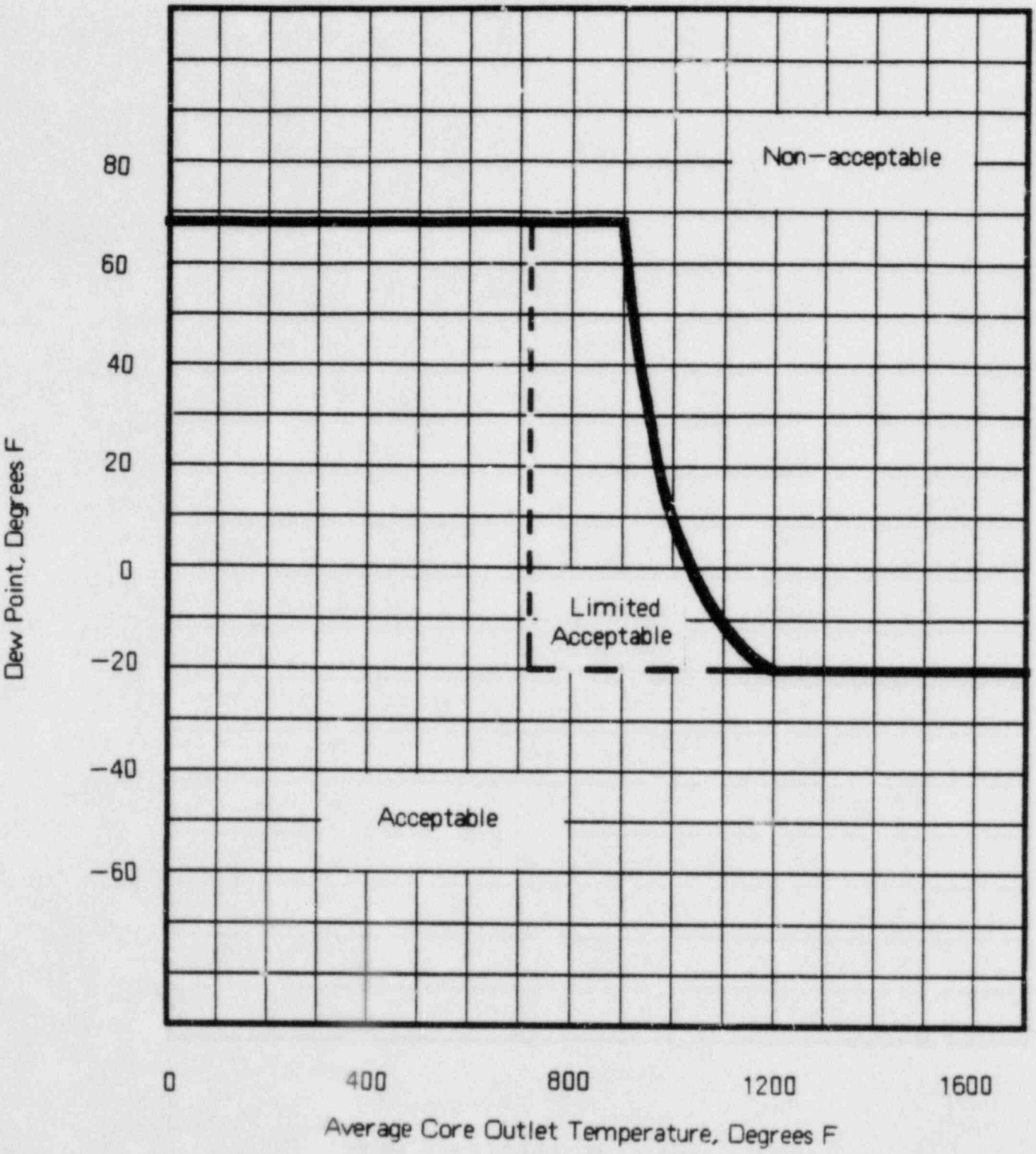
SURVEILLANCE REQUIREMENTS

4.4.3 When the reactor is critical, the primary coolant shall be analyzed for H2O, CO, and CO2:

a. At least once per 24 hours when the chemical impurity levels exceed 50 percent of the limits, or

b. At least once per 7 days at all other times.

DRAFT
APR 1 1985



LOOP IMPURITY LEVELS

Figure 3.4.3—1

DRAFT

MAR 1 1985

BASIS FOR SPECIFICATION LCO 3.4.3 / SR 4.4.3

During plant startup, CORE AVERAGE OUTLET TEMPERATURES will be below 1200 degrees F until the final stages when steam temperatures are increased to rated and the plant enters the power range. At these lower temperatures, graphite corrosion by the various chemical impurities is minimal and there is reduced concern for carbon transport.

There is a need, however, to prevent corrosion of metals in the primary coolant system and limit oxidation of burnable poison material in the core to acceptable levels.

In the presence of moisture, boron carbide B4C, is subject to oxidation at a temperature-dependent rate to form boron oxide, B203. In the event of subsequent significant steam inleakage the boron oxide is converted to volatile boric acid, which is capable of being steam-distilled from the core. Such an occurrence could produce an increase in core reactivity due to the loss of B-10. Taken in the context of the other constraints imposed by the presence of moisture in the primary coolant, it is only at CORE AVERAGE OUTLET TEMPERATURES above 725 degrees F that the rate of oxidation of boron carbide becomes sufficient to become a limiting parameter. At CORE AVERAGE OUTLET TEMPERATURES above 1200 degrees F, however, boron oxidation is of reduced significance because:

1) Moisture reaction with graphite significantly reduces the moisture concentration before it can react with the boron carbide, and,

2) Since the reactor must be at power to develop sustained temperatures in this range, the rate of B-10 depletion by burnup significantly exceeds the rate of B-4-C oxidation for allowable impurity levels (Specification 3.4.3).

The criterion used to establish the limits of Figure 3.4.3-1 in the range from 725 degrees F to 1200 degrees F was that not more than 10% of the beginning of life (BOL) Boron can be present as oxide over a refueling cycle. This criterion is based on the BOL Boron worth of 0.06 delta k, and the fact that 10% worth, 0.006 delta k, is substantially less than the minimum core shutdown margin of 0.014 delta k (FSAR Section 3.5.3.1), and only about one-half of the reactivity anomaly of 0.01 delta k specified in Specifications 3/4.1.7.

The stipulation used in developing the curve of Figure 3.4.3-1 is that with the CORE AVERAGE OUTLET TEMPERATURE in the range between 725 degrees F and 1200 degrees F, and with a primary coolant dew point temperature higher than -20 degrees F, operating time under these conditions would be limited to no more than 90 days over any refueling cycle. By combining this stipulation with the dew point limits shown in Figure 3.4.3-1 for this temperature range, the criterion that no more than 10% of the BOL B-10 will be present as oxide during a refueling cycle is met.

The dew point limit of 67 degrees F below 725 degrees F CORE AVERAGE OUTLET TEMPERATURE was selected to prevent corrosion of metal parts. It is an effective limit since all metal parts within the PCRV exposed to primary coolant are maintained at or above 75 degrees F.

The interval specified in Specification 4.4.2 will ensure that chemical inpurities will be measured and controlled in order to minimize corrosion of core materials.

DRAFT

APR 1 1985

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.1 HELIUM CIRCULATOR-POWER OPERATION AND LOW POWER

LIMITING CONDITION FOR OPERATION

3.5.1.1 At least one helium circulator in each loop shall be OPERABLE with:

a. Emergency feedwater circulator drive capable of providing the equivalent of 8000 rpm circulator speed at atmospheric pressure;

b. Two emergency water booster pumps (P-2109 and P-2110) OPERABLE, with the capacity to drive the circulator at 3% rated helium flow;

c. Two turbine water removal pumps (P-2103 and P-2103S) OPERABLE;

d. Two bearing water makeup pumps (P-2105 and P-2108) OPERABLE; and

e. The associated bearing water accumulator system OPERABLE.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION:

a. With at least one helium circulator but less than the above required helium circulators OPERABLE, restore the helium circulators to OPERABLE status within 72 hours or be in at least STARTUP within the next 12 hours.

b. With less than the above required OPERABLE equipment in b., c., or d. above, but with the capability to drive a circulator on steam notive power, restore a circulator to OPERABLE status within 7 days or be in at least STARTUP within the next 12 hours.

c. With all helium circulators inoperable and all forced circulation lost, be in SHUTDOWN within 15 minutes and restore forced circulation or depressurize the PCRV in accordance with the applicable requirement below:

1. As a function of reactor thermal power equal to or greater than 25% as given in Figure 3.5.1-1.

2. As a function of average core outlet temperature for thermal power levels less than 25% as given in Figure 3.5.1-2.

DRAFT

APR 1 1985

d. With a bearing water accumulator inoperable, restore it to OPERABLE status or remove the associated circulator(s) from service within 24 hours and declare the associated circulators inoperable.

SURVEILLANCE REQUIREMENT

4.5.1.1 The helium circulators shall be demonstrated OPERABLE:

a. At least one per 31 days by functionally testing the bearing water accumulator system which verifies accumulator flow to the circulator bearings at reduced pressure.

b. At least once per REFUELING CYCLE by:

1. Performing a turbine water removal pump (P-2103 and P-2103S) start test based on a simulated drain tank level to verify automatic actuator and pump start capability.

2. Performing a bearing water makeup pump (P-2105 and P-2108) start test based on a simulated low pressure in the backup bearing water supply line to verify automatic actuation and pump start capability.

3. Operation on water turbine drive by:

a. Verifying an equivalent 8000 rpm (at atmospheric pressure) on feedwater motive power using the emergency feedwater header.

b. Verifying an equivalent 3% rated helium flow on condensate at reduced pressure (to simulate firewater pump discharge) using both emergency water booster pumps (P-2109 and P-2110).

c. At least once per 10 years by inspection as follows:

1. A helium circulator compressor wheel rotor, turbine wheel and pelton wheel shall be inspected for both surface and subsurface defects in accordance with the appropriate methods, procedures, and associated acceptance criteria specified for Class I components in Article NB-2500, Section III, ASME Code. Other

DRAFT

APR 1 1985

helium circulator components, accessible without further disassembly than required to inspect these wheels, shall be visually examined.

2. At least 10% of primary coolant pressure boundary bolting and other structural bolting which has been removed for the inspection above and which is exposed to the primary coolant shall be nondestructively tested for identification of inherent or developed defects.

4. Monitoring the proper closure of the circulator helium shutoff valves.

DRAFT

APR 1 1985

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.1 HELIUM CIRCULATORS-STARTUP, SHUTDOWN AND REFUELING

LIMITING CONDITION FOR OPERATION

3.5.1.2 At least one helium circulator shall be OPERABLE with:

a. Boosted Firewater circulator water drive capable of providing an equivalent 3.0% rated helium flow;

b. Turbine Water Removal System with sufficient capacity to remove the water from one circulator water turbine;

c. One bearing water makeup pump OPERABLE;

d. The associated bearing water accumulator system OPERABLE.

APPLICABILITY: STARTUP, SHUTDOWN and REFUELING

ACTION:

a. With less than the above required OPERABLE equipment in a., b. or c. above, restore a circulator to OPERABLE status within 72 hours or be in SHUTDOWN within the next 24 hours.

b. With all helium circulators inoperable and all forced circulation lost, be in SHUTDOWN within 15 minutes and restore forced circulation or depressurize the PCRV in accordance with the applicalbe requirement below:

1) As a function of average core outlet temperature or thermal power levels less than 25% as given in Figure 3.5.1-2,

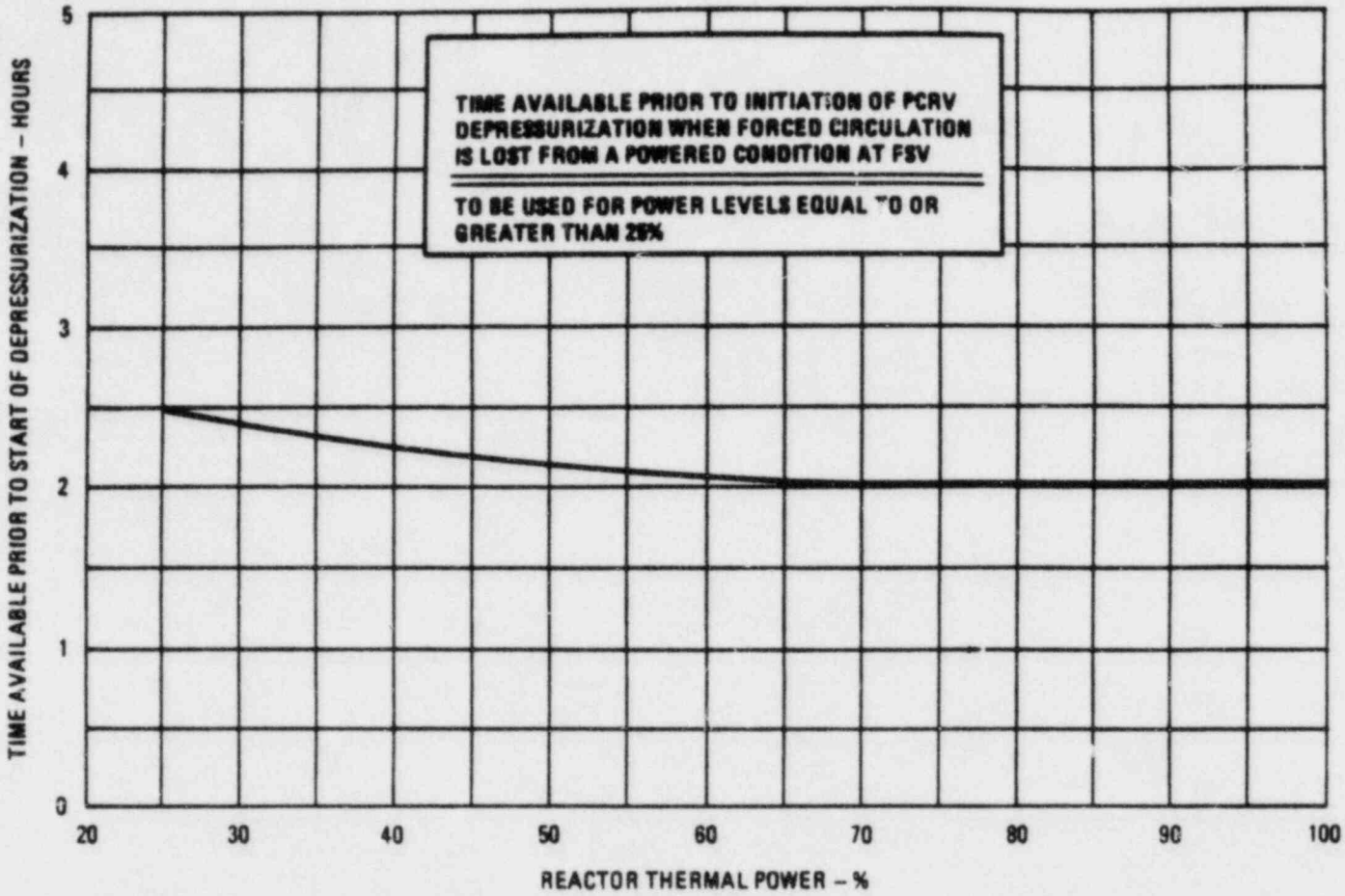
2) As a function of time from reactor shutdown as given in Figure 3.5.1-3.

SURVEILLANCE REQUIREMENT

4.5.1.2 No additional Surveillance Requirements beyond those specified in Specification 4.5.1.1.

DRAFT

APR 1 1985

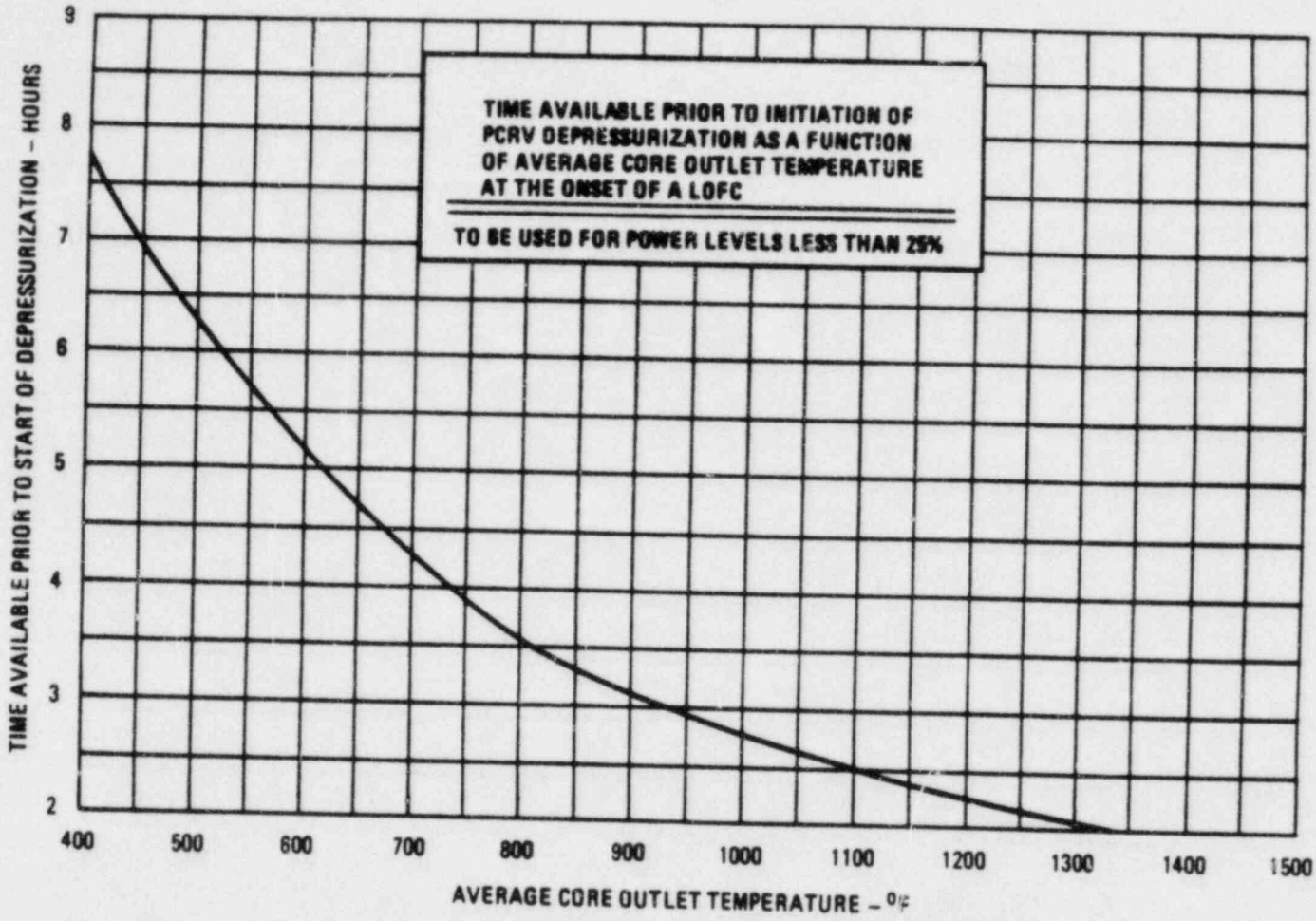


Time Available Prior to Initiation of PCRV Depressurization When Forced Circulation is Lost from a Powered Condition at FSV

Figure 3.5.1–1

DRAFT

APR 1 1985

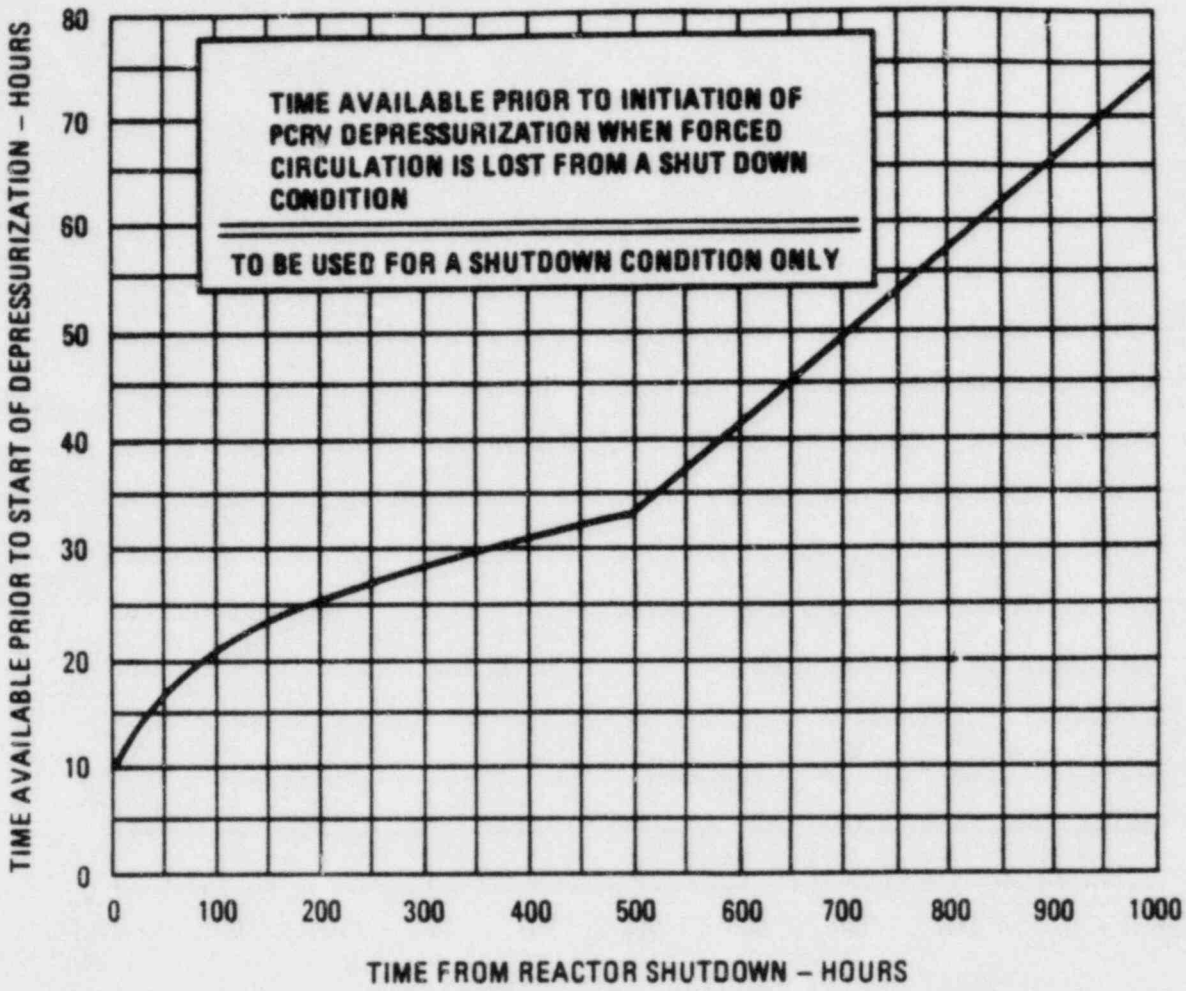


Time Available Prior to Initiation of PRCV Depressurization as a Function of Average Core Outlet Temperature at the Onset of a LOFC

Figure 3.5.1–2

DRAFT

APR 1 1985



Time Available Prior to Initiation of PCRV Depressurization When When Forced Circulation is Lost from a Shut Down Condition

Figure 3.5.1–3

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.5.1 / SR 4.5.1

One helium circulator, operating on water turbine drive via boosted firewater, provides for sufficient primary coolant circulation to assure safe shutdown cooling. With less than two emergency water booster pumps OPERABLE, coupled with the diverse and redundant means for circulator motive power (steam drive, feedwater and condensate) a 7 day action statement time is considered sufficient for restoration of the pumps. One circulator in each loop is specified to allow for a single failure in either the heat removal equipment or circulator auxiliary equipment which provides services to one loop.

Start of depressurization following loss of forced circulation is initiated as a function of prior power levels, with two (2) hours from full power operation being the most limiting case. Once depressurization is initiated, it is completed within 7 hours. Less limiting conditions, such as power operation at less than full power but greater than 25% of rated power, less than 25% of rated power or shutdown conditions permit additional time to the start of depressurization. These additional times are shown in Figures 3.5.1-1, 3.5.1-2, and 3.5.1-3 and are discussed in the FSAR Appendix D.

The capacity of each helium circulator water turbine drive method is discussed in FSAR Sections 14.4.2.1 and 14.4.2.2. Effective core cooling has been demonstrated analytically with each water turbine drive method. Further discussion of Safe Shutdown Cooling is included in the FSAR Section 10.3.9.

One turbine water removal pump has sufficient capacity to remove the water from two circulator water turbines. Additionally, a portable pump can be aligned to the turbine water removal system in the unlikely event that both turbine water removal pumps are inoperable. Also, the turbine water removal tank overflow to the reactor building sump will be used if the normal pump flowpath is lost. Therefore, a 7 day action statement time is considered sufficient for restoration of the pumps, based on the redundant and diverse means of removing water from the circulator water turbines.

Helium Circulator bearing water is normally supplied from the bearing water system and is backed up by the backup bearing water system supplied from the Emergency Feedwater Header. In the event of a failure in both of these systems, the water stored in the bearing water accumulators is adequate to safely shut down both helium circulators in a loop.

DRAFT
APR 1 1985

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.2 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.5.2.1 Two steam generators shall be OPERABLE with:

a. Both reheater and both economizer-evaporator-superheater (EES) sections OPERABLE, and

b. The steam generator superheater and reheater safety valves (V-2214, V-2215, V-2216, V-2245, V-2246, V-2247, V-2225 and V-2262) OPERABLE.

APPLICABILITY: POWER OPERATION AND LOW POWER

ACTION:

a. With at least one steam generator section but less than the above required steam generator sections OPERABLE, restore the required section(s) to OPERABLE status within 72 hours or be in STARTUP within the next 12 hours.

b. With all steam generator sections inoperable, be in SHUTDOWN within 15 minutes and restore at least one section to OPERABLE status or initiate depressurization in accordance with the times specified in Figures 3.5.1-1 and 3.5.1-2.

c. With the required safety valves inoperable, restore the valves to OPERABLE status within 72 hours, or restrict plant operation as follows:

1. With one EES safety valve inoperable, restrict plant operation to one boiler feed pump per operating loop.

SURVEILLANCE REQUIREMENTS

4.5.2.1 The steam generators shall be demonstrated OPERABLE:

a. At least once per 18 months by supplying flow through the emergency feedwater header and emergency condensate header to the steam generators as delineated per Specification 4.5.1.1.b.

DRAFT

APR 1 1985

b. At least once per five years by:

1. Testing the superheater and reheater safety valves and verifying their setpoints (Table 4.5.2-1).

2. The accessible portions of the following bimetallic welds shall be volumetrically examined for indications of subsurface defects:

   a) The main steam ring header collector to main steam piping weld for one steam generator module in each loop.

   b) The main steam ring header collector to collector drain piping weld for one steam generator module in each loop.

   c) The same two steam generator modules shall be re-examined at each interval.

   The initial examination shall be performed during SHUTDOWN or REFUELING prior to the beginning of Fuel Cycle 5. This initial examination shall also include the bimetallic welds described above for two additional steam generator modules in each loop.

c. Tube Leak Examination

Each time that a steam generator tube is plugged due to a leak, specimens from the accessible subheader tubes connected to the leaking inaccessible tubes(s) shall be metallographically examined.

The results of this metallographic examination shall be compared to the results from the specimens of all previous tube leaks.

A study shall be performed to evaluate the size and elevation of all tube leaks to determine if a cause of the leak or a trend in the degradation can be identified.

1. Acceptance Criteria

   An engineering evaluation shall be performed to determine the acceptability of:

   a) Any subsurface defects identified in Specification 4.5.2.1 b, and

   b) The results of the metallographic examination performed in Specification 4.5.2.1 c.

DRAFT

APR 1 1985

2. Reports

Within 30 days following the completion of each steam generator tube leak study per a Special Report shall be submitted to the Commission in accordance with Specification 6.9.2. This report shall include the estimated size and elevation of the leak(s), and the results of the metallographic and engineering analyses performed.

DRAFT

APR 1 1985

TABLE 4.5.2-1

STEAM GENERATOR SAFETY VALVES

VALVE NUMBER	LIFT SETTING
LOOP I	
V-2214	Less than or equal to 2917 psig
V-2215	Less than or equal to 2846 psig
V-2216	Less than or equal to 2774 psig
V-2225	Less than or equal to 1133 psig
LOOP II	
V-2245	Less than or equal to 2917 psig
V-2246	Less than or equal to 2846 psig
V-2247	Less than or equal to 2774 psig
V-2262	Less than or equal to 1133 psig

DRAFT

APR 1 1985

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.2 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.5.2.2 The steam generators shall be OPERABLE with:

a. At least two sections (reheater/economizer-evaporator-superheater) in any combination of one or both steam generators OPERABLE, and

b. The steam generator superheater and reheater safety valves (V-2214, V-2215, V-2216, V-2245, V-2246, V-2247, V-2225 and V-2262) which protect the operating sections of the steam generators shall be OPERABLE.

APPLICABILITY: STARTUP

ACTION:

a. With at least one steam generator section but less than the above required steam generator section(s) OPERABLE, restore the required section(s) to OPERABLE status within 72 hours or be in SHUTDOWN within the next 12 hours.

b. With all steam generator sections inoperable, be in SHUTDOWN within 15 minutes and restore one section to OPERABLE status or initiate PCRV depressurization in accordance with the times specified in Figures 3.5.1-2 and 3.5.1-3.

c. With the required safety valves inoperable, restore the valves to OPERABLE status within 72 hours, or restrict plant operation as follows:

1. With one EES safety valve inoperable, restrict plant operation to one boiler feed pump per operating loop.

SURVEILLANCE REQUIREMENTS

4.5.2.2 No additional surveillances required other than those identified per Specification 4.5.2.1.

DRAFT

APR 1 1985

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.2 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.5.2.3 a. At least the reheater section or the economizer-evaporator-superheater section of one steam generator shall be OPERABLE, and

b. The steam generator superheater or reheater safety valve(s) which protect the operating section of the steam generator shall be OPERABLE.

APPLICABILITY: SHUTDOWN AND REFUELING

ACTION:

a. With the above required steam generator section OPERABLE, restore it to inoperable status prior to the time calculated for the core to heatup from decay heat to a calculated core average temperature of 760 degrees F or initiate PCRV depressurization in accordance with the time specified in Figure 3.5.1-3.

b. With the required safety valves inoperable, restore the valves to OPERABLE status within 72 hours, or restrict plant operation as follows:

1. With one EES safety valve inoperable, restrict plant operation to one boiler feed pump per operating loop.

SURVEILLANCE REQUIREMENTS

4.5.2.3 No additional surveillance requirements other than those identified per Specification 4.5.2.1.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.5.2, / SR 4.5.2

The requirements for OPERABLE steam generators provide an adequate means for removing heat from the primary reactor coolant system to the secondary reactor coolant system. During POWER OPERATION and LOW POWER, all steam generator sections are required for plant operation. This ensures safe shutdown cooling capability for those transients identified in Chapter 14 of the FSAR.

During STARTUP, any two steam generator sections are required. For SHUTDOWN purposes, either the reheater section or the economizer-evaporator-superheater section of one steam generator can be used for shutdown heat removal from the primary coolant. In addition to the normal water sources, the reheater or EES sections can receive water from either the associated emergency condensate header or the emergency feedwater header.

The economizer-evaporator-superheater section of each steam generator loop is protected by three spring-loaded safety valves, each with one-third nominal capacity. The reheater section of each steam generator loop is protected from over pressure transients by a single safety valve. These steam safety valves are described in FSAR Section 10.2.5.3.

During ALL MODES, with one safety valve inoperable, plant operation is restricted to a condition for which the remaining safety valves have sufficient relieving capability to prevent overpressurization of the steam generator section(s).

Bimetallic Weld Examination

The steam generator crossover tube bimetallic welds between Incoloy 800 and 2 1/4 Cr-1 Mo materials are not accessible for examination. The bimetallic welds between the steam generator ring header collector, the main steam piping, and the collector drain piping are accessible, involve the same materials, and operate at conditions not significantly different from the crossover tube bimetallic welds. The collector drain piping weld is also geometrically similar to the crossover tube weld. Although minimal degradation is expected to occur, this specification allows for detection of defects which might result from conditions that can uniquely affect bimetallic welds made between these materials. Additional collector welds are inspected at the initial examination to establish a baseline which could be used, should defects be found in later inspections and additional examinations subsequently be required.

DRAFT

APR 1 1985

Tube Leak Examination

During the lifetime of the plant, a certain number of steam generator tube leaks are expected to occur, and the steam generators have been designed to have these leaking tube subheaders plugged without affecting the plant's performance as shown in FSAR Table 4.2-5. The consequences of steam generator tube leaks have been analyzed in FSAR Section 14.5.

It is important to identify the size and elevation of steam generator tube leaks and to metallographically examine the subheader tube material because this information can be used to analyze any trend or generic cause of tube leaks. Conclusive identification of the cause of a steam generator tube leak may enable modifications and/or changes in operation to increase the reliability and life of the steam generators and to prevent a quantity of tube failures in excess of those analyzed in the FSAR.

Because of the subheader designs leading to the steam generator tube bundles, in-situ internal or external inspection and evaluation of a tube leak to establish a conclusive cause is not practical. Metallographic examination of the accessible connecting subheader tube will show the condition of the internal subheader wall, giving an indication of the conditions of the leaking tube internal wall, thereby demonstrating the effectiveness of water chemistry controls. Determining the approximate size and elevation of the tube leak may enable evaluation of other possible leak causes such as tube/tube support plate interface effects.

The surveillance plan outlined above is considered adequate to evaluate steam generator tube integrity and assure that the consequences of postulated tube leaks remain within the limits analyzed in the FSAR.

DRAFT

APR 1 1985

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.3 EMERGENCY CONDENSATE AND EMERGENCY FEEDWATER HEADERS

LIMITING CONDITION FOR OPERATION

3.5.3.1 The emergency condensate header and the emergency feedwater header shall be OPERABLE.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION:

a. With either the emergency condensate header or the emergency feedwater header inoperable, restore the inoperable header to OPERABLE status within 72 hours or be in STARTUP within the next 12 hours.

b. With both the emergency condensate header and the emergency feedwater header inoperable, restore at least one of the headers to OPERABLE status within 24 hours and complete ACTION a. above, or be in STARTUP within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.1 No additional surveillance requirements are required other than those surveillances identified in Specification 4.5.2.1 a.

DRAFT

APR 1 1985

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.3 EMERGENCY CONDENSATE AND EMERGENCY FEEDWATER HEADERS

LIMITING CONDITION FOR OPERATION

3.5.3.2 The emergency condensate header or the emergency feedwater header shall be OPERABLE.

APPLICABILITY: STARTUP

ACTION: With the emergency condensate header and the emergency feedwater header inoperable, restore one header to OPERABLE status within 72 hours or be in SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.2 No additional surveillance requirements are required other than those surveillances identified in Specification 4.5.2.1 a.

DRAFT

APR 1 1985

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.3 EMERGENCY CONDENSATE AND EMERGENCY FEEDWATER HEADERS

LIMITING CONDITION FOR OPERATION

3.5.3.3 Either the emergency feedwater header or the emergency condensate shall be OPERABLE.

APPLICABILITY: SHUTDOWN and REFUELING

ACTION: With less than the above required emergency feedwater or emergency condensate headers OPERABLE, restore at least one header to OPERABLE status prior to the time calculated for the core to heatup from decay heat to a calculated core average temperature of 760 degrees F.

SURVEILLANCE REQUIREMENTS

4.5.3.3 No additional surveillance requirements are required other than those surveillances identified in Specification 4.5.2.1 a.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.5.3 / SR 4.5.3

The OPERABILITY of the emergency condensate and emergency feedwater headers ensures redundant water supply paths to the helium circulators and steam generators for safe shutdown of the plant. In the event of a failure of the normal feedwater line, the availability of either the emergency feedwater or emergency condensate lines provides adequate shutdown capability.

DRAFT
APR 1 1985

PRIMARY COOLANT SYSTEM

3/4.5.4 FIREWATER SUPPLY SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.4 The firewater supply system shall be OPERABLE with:

a. At least two OPERABLE circulating water makeup pumps (P4118-P, P4118S-P, and P4118SX-P) connectible to an essential bus,

b. Two OPERABLE circulating water makeup headers (main and emergency),

c. Two OPERABLE firewater pumps, both the motor driven (P-4501) and the engine driven (P-4501S), including the associated pump pits and for the engine driven pump, at least 370 gallons of fuel in the day tank,

d. An inventory of at least 20 million gallons in the circulating water makeup storage ponds, and

e. An OPERABLE flow path from the discharge of the firewater pumps to isolation valves for the emergency condensate header.

APPLICABILITY: ALL MODES

ACTION:

a. With less than the requirements specified in a through d above, restore the inoperable equipment to OPERABLE status within 24 hours or be in SHUTDOWN within the next 24 hours.

b. With the flow path from the discharge of the firewater pumps to the isolation valves for the emergency condensate header inoperable, restore it to OPERABLE status within 1 hour or be in SHUTDOWN within the next 24 hours.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.5.4 The firewater supply system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the inventory in the circulating water makeup ponds is at least 20 million gallons.

b. At least once per 7 days by:

1. Verifying that the electrolyte level of each diesel starting battery is above the plates, and

2. Verifying that the overall battery voltage is greater than or equal to 24 volts.

c. At least once per 31 days by:

1. Verifying the fuel storage tank contains at least 370 gallons of fuel.

2. Verifying the diesel starts and operates for at least 30 minutes on recirculation flow.

3. Starting the electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

4. Verifying each valve that is not locked, sealed or otherwise secured in place is in its correct position.

d. At least once per 92 days by:

1. Verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975 is within the acceptance limits specified in Table 1 of ASTM D975-1977 when checked for viscosity, water and sediment.

2. Verifying that the specific gravity is appropriate for continued service of each battery.

e. At least once per 12 months by performance of a system flush.

DRAFT

APR 1 1985

f. At least once per 18 months by:

1. Cycling each testable valve in the flowpath through at least one complete cycle of full travel.

2. Performing a CHANNEL CALIBRATION on the pond level instrumentation and on the circulating water makeup pump controls and instrumentation, including the fire water pump pits.

3. Verifying that the batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

4. Assuring that the battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

5. Performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence.

g. At least once per 18 months, during SHUTDOWN, by subjecting the diesel to an inspection in accordance with procedures prepared in conjuction with its manufacturer's recommendations for the class of service.

h. At least once per 3 years by performing a flow test.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.5.4 / SR 4.5.4

The OPERABILITY of the firewater supply system ensures that adequate firewater is available to the fire suppression system. The firewater suppression system consists of spray and/or sprinklers, fire hose stations, and yard fire hydrants. The collective capability of the firewater suppression systems is adequate to minimized potential damage to safety related equipment.

In addition to the fire suppression function, either of the firewater pumps operating in conjunction with either emergency water booster pump provides adequate capacity to operate a circulator water turbine and supply emergency cooling water to the steam generators for safe shutdown cooling. With the 370 gallons of fuel in storage, the engine driven fire pump can operate at rated conditions for 24 hours which is adequate time to have more fuel delivered to the site.

The circulating water makeup system provides at least 20 million gallons of water to the service water and firewater systems. During extremely cold weather, formation of ice on the surface of the circulating water storage ponds can occur. However, the ice formation does not change the LCO limit for at least 20 million gallons of water. Safe Shutdown Cooling requirements are discussed in Sections 1.4, 10.3 and 14.4 of the FSAR.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

PCRV SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.6.1.1 The PCRV safety valve installation shall be OPERABLE with:

a. Rupture disc and safety valve settings as follows:

1. Rupture Disc (Low Set Safety Valve).....812 psig plus or minus 8 psig.

2. Low Set Safety Valve.....796 psig plus or minus 8 psig.

3. Rupture Disc (High Set Safety Valve).....832 psig plus or minus 8 psig.

4. High Set Safety Valve.....812 psig plus or minus 8 psig.

b. Both inlet block valves locked open,

c. Less than 5 psig between each rupture disc and relief valve,

d. Less than 16.75 psig in the safety valve containment tank, and

e. Less than 195 psig in the safety valve pilot stage bellows.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia.

ACTION: With the PCRV safety valve installation inoperable, restore it to OPERABLE status within 12 hours or be in at least SHUTDOWN within the next 24 hours and depressurized to less than 100 psia within the following 30 hours.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.6.1.1 The PCRV safety valve installation shall be demonstrated OPERABLE:

a. At least once per 31 days by functionally testing the pressure switch and alarm for each interspace between a rupture disc and the corresponding PCRV safety valve.

b. At least once per 12 months by:

1. Functionally testing and calibrating the pressure switch and alarm for the PCRV safety valve containment tank, and

2. Calibrating the pressure switch and alarm for each interspace between a rupture disc and the corresponding PCRV safety valve.

c. During each refueling outage by testing one of the PCRV safety valve and rupture disc assemblies, on an alternating basis, with the interval between testing individual assemblies not to exceed 5 years.

d. When either PCRV safety valve and rupture disc assembly is tested by:

1. Functionally testing and calibrating the position indication circuits and alarms associated with the PCRV safety valve and rupture disc assembly block valves, and

2. Functionally testing and calibrating the associated pressure switch and alarm for the safety valve bellows.

DRAFT

APR 1 1985

BASIS FOR LCO 3.6.1.1 / SR 4.6.1.1

The PCRV safety valve installation (consisting of a containment tank containing two parallel systems, each of which has a manual block valve and a rupture disc upstream of the safety valve and which discharge to the atmosphere via a single particulate filter) provides the ultimate protection against overpressurization of the PCRV. Although each train is capable of protecting the PCRV from overpressure by itself, both trains are required to assure protection assuming a single failure. In order for the installation to be considered OPERABLE, the rupture discs and safety valves must be set as specified in Specification 2.2, the block valves must be open and system pressures must be within the specified limits. The pressure limits are based on maximum alarm setpoints including calibration tolerances.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

STEAM GENERATOR/CIRCULATOR PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.6.1.2 One PCRV penetration overpressure protection train protecting each group of steam generator penetrations and each circulator penetration shall be OPERABLE with:

a. Rupture disc and safety valve settings as follows:

1. Helium Circulator Penetration

Rupture Disc .....825 psig plus or minus 2%

Safety Valve .....805 psig plus or minus 3%

2. Steam Generator Penetration

Rupture Disc .....825 psig plus or minus 2%

Safety Valve .....475 psig plus or minus 3%

b. The associated block valve open, and

c. Less than 5 psig between the rupture disc and relief valve.

APPLICABILITY: At all times

ACTION: With both PCRV penetration overpressure protection trains protecting any group of steam generator or circulator penetrations inoperable, restore one train to OPERABLE status or depressurize the process piping in the affected penetrations to less than primary coolant pressure within 12 hours or be in at least SHUTDOWN within the next 24 hours and depressurized to less than 100 psia within the following 30 hours.

DRAFT

SURVEILLANCE REQUIREMENTS APR 1 1985

4.6.1.2 The steam generator and circulator penetration overpressure protection trains shall be demonstrated OPERABLE:

a. At least once per 31 days by functionally testing the pressure switch and alarm for each interspace between a rupture disc and the corresponding safety valve.

b. At least once per 12 months by calibrating the pressure switch and alarm for each interspace between a rupture disc and the corresponding safety valve.

c. At least once per 5 calendar years, on a STAGGERED TEST BASIS, by:

1. Testing each of the two redundant assemblies so that one safety valve and one rupture disc for each penetration interspace is tested at an approximate interval of two and one half years.

2. Functionally testing the control, interlock, and indication circuits associated with each of the penetration overpressure protection assembly shutoff valves.

DRAFT

APR 1 1985

BASIS FOR LCO 3.6.1.2 / SR 4.6.1.2

The steam generator and circulator penetrations are the only PCRV penetrations that contain process fluids at pressures high enough to necessitate overpressure protection for the penetrations. Separate overpressure protection trains are provided for a) the six steam generator module penetrations of each loop, and b) each of the four helium circulator penetrations. Each train consists of two parallel relief assemblies consisting of a manual block valve, a rupture disc, and a safety valve. The block valves in a train are interlocked so that only one valve can be opened at any time. In order for the train to be considered OPERABLE, the rupture disc and safety valve must be set as specified in Specification 2.3, the block valve must be open and the interspace between each rupture disc and safety valve must not be pressurized above 5 psig.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

INTERSPACE PRESSURIZATION

LIMITING CONDITION FOR OPERATION

3.6.1.3 The interspaces between the primary and secondary PCRV penetration closures shall be maintained at a pressure greater than primary coolant pressure with purified helium gas except, if a leak pathway from one or more steam generator interspace(s) to the cold reheat steam system exists, the steam generator penetration interspaces for the affected loop(s) may be maintained at a pressure less than primary coolant pressure, but greater than cold reheat steam pressure.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia

ACTION: With one or more penetration interspaces below the minimum required pressure, restore the necessary pressurization within 12 hours or be in at least SHUTDOWN within the next 24 hours and depressurized to less than 100 psia within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 The required pressurization of the PCRV penetration interspaces shall be demonstrated:

a. At least once every 24 hours by verifying that the PCRV penetration interspace supply pressure is within its limits,

b. At least once per 31 days by functional testing of the instrumentation and valves that control and monitor the pressure of the purified helium supply to the PCRV penetration interspaces, and

c. At least once per 12 months by calibrating the instrumentation that controls and monitors the pressure of the purified helium supply to the PCRV penetration interspaces.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.6.1.3 / SR 4.6.1.3

Pressurization of PCRV penetration interspaces to a pressure above primary coolant pressure 1) positively prevents any leakage of contaminated helium from the primary coolant system through the primary closure, and 2) assures that any leakage through the secondary closure into the Reactor Building will be purified helium and thus will have no radiological consequences.

An exception to the above is permitted for the steam generator penetration interspaces because of the potential leakage of purified helium gas from those interspaces into the cold reheat steam system piping within the interspaces. In that event, the steam generator penetration interspaces in the affected loop may be maintained at a pressure greater than cold reheat steam pressure, but less than primary coolant pressure. This mode of operation reduces the driving force for leakage and significantly reduces the leakage of purified helium gas to the cold reheat system which, in turn, allows the maintenance of condenser vacuum required for normal plant power operation. In this mode of operation, there exists the potential for contaminated helium gas leakage through the primary closure and into the reheat steam system where it would be stripped from the steam by the main condenser air ejector and ultimately exhausted to atmosphere through the monitored plant exhaust stack. Consequently, more stringent leak tightness and radiation monitoring requirements are imposed for this mode of operation by Specifications 3.6.1.4 and 3.6.1.5, to provide compensatory measures that demonstrate the integrity of the primary closure and prevent the leakage of unacceptable quantities of contaminated helium gas. Condenser air ejector operability is assured by Specification 8.1.1.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

INTERSPACE LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.4 The helium leakage through PCRV penetration closures shall be maintained within the following limits:

a. Leakage through all the primary closure seals in any penetration group shall not exceed an equivalent leak rate of 400 pounds per day at a differential pressure of 10 psi, and

b. Leakage through all the secondary closure seals combined shall not exceed an equivalent leak rate of 400 pounds per day at a differential pressure of 688 psi, and

c. Leakage through all the primary closure seals in the steam generator penetration group(s) pressurized to less than primary coolant pressure shall not result in the release of greater than 1.4 curies per day, and

d. Leakage from any steam generator penetration group to the reheat steam system shall not exceed 700 pounds per day.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia.

ACTION: With interspace helium leakage exceeding the above limits, reduce the helium leakage to meet the limits within 24 hours or be in at least SHUTDOWN within the next 24 hours and depressurized to less than 100 psia within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The surveillance of the PCRV penetration interspace leakage shall be as follows:

a. At least once per 31 days, the instrumentation monitoring PCRV penetration interspace pressurization gas flows, including alarms and high flow isolation shall be functionally tested.

DRAFT

APR 1 1985

b. At least once per 92 days or within 12 hours after an unanticipated increase in pressurization gas flow is alarmed, PCRV penetration primary and secondary closure leakage and steam generator penetration interspace leakage to the reheat steam system shall be determined.

c. At least once per 12 months, the instrumentation monitoring PCRV penetration interspace pressurization gas flows, including alarms and high flow isolation shall be calibrated.

d. Within 12 hours after steam generator penetration interspace activity is detected, and once per 24 hours thereafter, determine the release rate of primary coolant activity via the condenser air ejector exhaust.

DRAFT

APR 1 1985

BASIS FOR LCO 3.6.1.4 / SR 4.6.1.4

PCRV penetration interspaces are normally maintained at a pressure greater than the primary coolant pressure by supplying them with clean helium from either the high pressure helium storage tanks or from the helium purification system so that any leakage through either the primary or secondary closure seals will be clean helium.

The leakage rate limitations for the primary closures are based on a permissible leakage rate of 1145 pounds per hour at a differential pressure of 688 psig, which would be the differential pressure across a primary closure in the event a secondary closure should fail. Converting the 1145 pounds per hour leakage rate to normal operating conditions of 10 psi differential pressure indicates an operating limiting leakage rate of 400 pounds per day (16.7 pounds per hour).

Secondary seal leakage during normal operation is leakage of purified helium. The secondary seal leakage is limited to 400 pounds per day at the normal operating differential pressure of 688 psi to assure that the penetration interspaces can be maintained at the pressures required by Specification 3.6.1.3.

Because of a potential leak pathway between the steam generator penetrations and the cold reheat steam system piping within the penetration, the steam generator penetration interspaces may be operated at a pressure below primary coolant pressure, but above cold reheat steam pressure. When operating in this mode, there exists a potential effluent pathway of primary coolant leakage across the primary closure and into the reheat steam system. The helium, plus primarily noble gases, would be removed by the condenser air ejector and exhausted out the plant stack. The release of noble gases via this pathway is limited to 10% of the design objective for the plant's radioactive gas releases of 4160 curies per year. This results in an allowable daily release of 1.4 curies based on a plant capacity factor of 0.80 (292 days operation per year).

The allowed leakage of 700 pounds per day of purified helium to the reheat steam system is based on maintenance of satisfactory main condenser vacuum and economic considerations. Since this leakage is purified helium, there is no radiological impact. Monitoring by the air ejector radiation monitor and the reheat loop activity monitor(s) provides assurance of the absence of radioactive effluent.

DRAFT

APR 1 1985

In the determination of closure leakage at the reference differential pressure, laminar leakage flow shall be conservatively assumed, therefore in correcting the determined closure leakage to reference differential pressure, the ratio of the reference differential pressure and test differential pressure shall be used.

In the determination of the primary coolant release rate (curie per day leak rate) the flow rate of the condenser air ejectors shall be assumed to be their maximum capacity of 15 cfm for each operating air ejector.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

INTERSPACE RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.6.1.5 With either or both steam generator penetration interspace groups being maintained above cold reheat pressure, but below primary coolant pressure:

a. The associated reheat loop activity monitor channel 2263 and/or 2264 shall be OPERABLE,

b. The affected steam generator interspace group(s) shall be monitored for increases in gross activity by activity monitor channel 2263 or 2264, and

c. Gross activity shall not exceed 200% of the background value.

APPLICABILITY: Above 100 psia PCRV pressurization when either or both steam generator penetration interspace groups are being maintained above cold reheat pressure, but below primary coolant pressure.

ACTION:

a. With a required reheat loop activity monitor channel 2263 or 2264 inoperable:

1. Take a grab sample of the affected steam generator loop penetration interspaces at least once every eight (8) hours and analyze the samples for noble gas gross activity within 24 hours, and

2. Restore the required channel to OPERABLE status within 7 days or be in at least SHUTDOWN within the next 24 hours and depressurized to less than 100 psia within the following 30 hours.

DRAFT

APR 1 1985

b. With steam generator penetration interspace gross activity exceeding 200% of the background value, determine that the leak tightness of the steam generator penetration interspace primary closures meets the above requirements within the next 24 hours or be at least SHUTDOWN within the subsequent 24 hours and depressurized to less than 100 psia within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 Reheat loop activity monitor channels 2263 and/or 2264 shall be demonstrated OPERABLE when the associated steam generator penetration interspace group III or IV is being maintained above cold reheat steam pressure, but below primary coolant pressure:

a. At least once per 31 days by:

1. Performance of a CHANNEL FUNCTIONAL TEST, and

2. Verifying that each valve (manual, power operated, or automatic) in the sample flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months by:

1. Performance of a CHANNEL CALIBRATION and,

2. Cycling each power operated valve in the sample flow path through at least one complete cycle of full travel.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.6.1.5 / SR 4.6.1.5

Monitoring of the steam generator penetration interspace gas activity with radiation monitoring channels 2263 and/or 2264 provides a means of verifying primary closure leak tightness while operating with reduced steam generator penetration interspace pressure.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.2 PCRV LINER COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2 a. The PCRV Liner Cooling System shall be OPERABLE with:

1. Two PCRV Liner Cooling System loops OPERATING (one heat exchanger and one pump in each loop in service),

2. At least three (3) out of any four (4) PCRV Liner Cooling System tubes on the liner side walls OPERABLE,

3. At least five (5) out of any six (6) PCRV Liner Cooling System tubes on the liner top head or core support floor top casing OPERABLE, and

4. PCRV Liner Cooling tubes adjacent to a failed tube shall be OPERABLE,

b. If there is only one OPERATING loop, at least two heat exchangers and one pump shall be in service.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION:

a. With only one PCRV Liner Cooling System loop OPERATING, restore the second loop to OPERATING within 48 hours or be in STARTUP within the subsequent 12 hours and be in SHUTDOWN within the following 12 hours.

b. With less than the above required number of PCRV Liner Cooling tubes OPERABLE, restore the required PCRV Liner Cooling System tubes to OPERABLE within 24 hours or be in STARTUP within the subsequent 12 hours and be in SHUTDOWN within the following 12 hours.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.6.2 The PCRV Liner Cooling System shall be demonstrated OPERABLE:

a. At least once per 24 hours, by verifying that each PCRV Liner Cooling System loop is circulating cooling water at a flow rate greater than 1100 gpm.

b. At least once per 31 days, by verifying at least every 31 DAYS that liner cooling tube thermocouple outputs and their respective inlet header temperatures are within 20 degrees F (for an operating loop) with the following exceptions:

Tube	Thermocouple Element No.	Scanner Channel
to be	46205-40	401
supplied	46205-36	411
later.	46203-1	21
	46203-2	321

The temperature limit for the tubes specifically listed above is 40 degrees F (for an OPERATING loop).

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.6.2 / SR 4.6.2

The limitations on the number of OPERATING PCRV Liner Cooling System loops and failed tubes ensures that PCRV Liner Cooling System temperatures and thermal stresses within the concrete will remain within the FSAR design limits (FSAR Section 5.9.2). Analytical calculations in support of the PCRV Liner Cooling System design (FSAR Section 5.9.2.4) demonstrate that operation at full power with one cooling loop for 48 hours satisfies the criterion which specifies a maximum temperature increase of 20 degrees F in the bulk of the PCRV concrete. Operation on one loop during a loss-of-forced-circulation accident using the alternate cooling method with increased cooling water system cover pressure of 30 psig may result in temperature rises across individual cooling tubes of 240 degrees F (outlet temperature of 340 degrees F). These conditions result in acceptable liner cooling for this analyzed condition and PCRV structural integrity is preserved (FSAR Section D.1.2.1.5).

PCRV liner cooling tube failures, whether the result of leakage or blocking, do not affect the integrity of the PCRV as long as such a failure is limited to a single tube in any set of four on the side walls, or a single tube in any set of six on the PCRV liner top head and core support floor top casing. These requirements also preclude operation with two adjacent tubes failed. In these cases, the local temperature in the concrete would be less than 250 degrees F, an allowable and acceptable concrete temperature (FSAR 5.9.2.4).

Operation of the PCRV Liner Cooling System during startup testing demonstrated hot spots on the liner. These locations were identified and analyzed in the above FSAR Sections. The analysis indicates that operation with the known hot spots does not compromise PCRV integrity, thus continued operation is acceptable. The temperature limits of the tubes associated with the hot spots are specified separately. A 40 degree F temperature rise across these tubes was observed during plant startup and factored into anaylses of the hot spots.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.3 PCRV LINER COOLING SYSTEM TEMPERATURES

LIMITING CONDITIONS FOR OPERATION

3.6.3 The PCRV Liner Cooling System Temperatures shall be maintained within the following limits:

a. The maximum average temperature difference between the common PCRV cooling water discharge temperature and the PCRV external concrete surface temperature shall not exceed 50 degrees F.

b. The maximum PCRV Liner Cooling System water temperature shall not exceed 120 degrees F.

c. The maximum change of the PCRV concrete temperature shall not exceed 14 degrees F per week.

d. The maximum temperature difference across the PCRV Liner Cooling Water Heat Exchanger shall not exceed 20 degrees F.

e. The minimum average water temperature of the PCRV Liner Cooling System Heat Exchanger shall be greater than or equal to 100 degrees F.

APPLICABILITY: At all times

ACTION:

a. If any of the above conditions are not met, take immediate corrective action to restore the PCRV Liner Cooling System temperatures within the limits stated above within 24 hours or be in STARTUP within the next 12 hours and be in SHUTDOWN within the next 12 hours.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.6.3 The PCRV Liner Cooling System Temperatures shall be demonstrated to be within their respective limits at least once per 24 hours by:

a. Verifying that the maximum temperature difference averaged over a 24 hour period between the PCRV external concrete surface temperature and the common PCRV cooling water discharge temperature in each loop does not exceed 50 degrees F.

b. Verifying that the maximum PCRV Liner Cooling Water Outlet temperature does not exceed 120 degrees F as measured by PCRV Liner Cooling Water Outlet Temperature in each loop.

c. Verifying that the change in PCRV concrete temperature does not exceed 14 degrees F per week as indicated by the weekly average water temperature measured at the common PCRV cooling water outlet temperature in each loop.

d. Verifying that the maximum delta T across the PCRV Liner Cooling System Heat Exchanger does not exceed 20 degrees F as measured by the PCRV heat exchanger outlet temperature and the common PCRV Liner Cooling Water Outlet temperature in each loop.

e. Verifying that the minimum water temperature of the PCRV Liner Cooling System is greater than or equal to 100 degrees F as measured by the average of the PCRV Liner Cooling System heat exchanger inlet and outlet temperatures.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.6.3/ SR 4.6.3

FSAR Section 5.7, 5.9, 5.12 and 9.7 provide bases for the limits incorporated into Specification 3.6.3. The PCRV Liner and its associated cooling system assists in maintaining the integrity of the PCRV concrete.

PCRV bulk concrete temperature is not measured directly. The PCRV Liner Cooling System Temperatures and their specified frequency of measurement ensures that thermal stresses on the PCRV concrete and liner are within FSAR analyses described above and that PCRV integrity is maintained.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.4 PCRV INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.4 The structural integrity of the PCRV shall be maintained at a level consistent with the acceptance criteria specified below.

APPLICABILITY: At all times

ACTION: With the structural integrity of the PCRV not conforming to the acceptance criteria, restore the structural integrity to within the limits within 24 hours or be SHUTDOWN within the subsequent 6 hours and depressurized to below 100 psia within the next 30 hours.

SURVEILLANCE REQUIREMENT

4.6.4.1 The structural integrity of the PCRV shall be demonstrated by performance of the following surveillance requirements:

a. At least once per 6 months:

1. Visually examine the anchor assemblies as shown in Table 4.6.4-1 including tendon wires, button heads, anchor/bushing assemblies, stressing washers, shims, bearing plates, and surrounding concrete, and

2. Perform lift off tests to determine the load being carried by tendons for a control group identified in Table 4.6.4-2.

b. At least once per 18 months perform lift off tests to determine the load being carried by the new tendons identified in Table 4.6.4-2.

DRAFT

APR 1 1985

c. Acceptance Criteria:

Engineering evaluations of the above test results will be made as the tendon inspection/testing program progresses with the intent of ensuring that PCRV integrity is maintained. Specific engineering evaluations will be mandatory for any circumferential barrel tendon with greater than or equal to 15% failed wires* and for any tendon in any of the remaining tendon groups with greater than or equal to 20% failed wires*.

A special report, summarizing the results of this interim inspection/testing program, shall be submitted to the Commission at the conclusion of the program.

*Note: Failed wires shall be defined as 1) those having raised button heads in the anchor assembly, 2) those previously identified as failed by visual inspection, and 3) for tendons which are not accessible on both ends, 20% of the number of failed wires identified on the accessible end.

TABLE 4.6.4-1

TENDON VISUAL INSPECTION PROGRAM

TENDON GROUPS	TOTAL NUMBER OF TENDONS (1)	TOTAL NUMBER OF NEW TENDONS (2)	TOTAL NUMBER OF CONTROL TENDONS (3)
Circumferential	16	13	3
Top Cross Head	2	1	1
Bottom Cross Head	8	6	2
Longitudinal	30	24	6

NOTES:

1. The total number of tendon anchor assemblies to be inspected shall correspond to the indicated number of tendons in that group. With the exception of longitudinal tendons, both ends of all designated tendons shall be inspected, if accessible. Longitudinal tendons shall be inspected only from the top end.

2. The total number of new tendons consists of a tendon population selected at random for examination or testing over the next specified surveillance period. Selection shall be such that the total population of accessible tendons in that group shall be inspected/tested before beginning any repeat inspection/tests.

3. The total number of control tendons consists of the same population of tendons in each tendon group that will be selected and will remain constant for all inspection/test surveillance cycles. The criteria for selection of these tendons shall be to select those tendons which represent conditions in which corrosion is most pronounced tempered by ready accessibility.

DRAFT

APR 1 1985

TABLE 4.6.4-2

TENDON LIFTOFF PROGRAM

TENDON GROUPS	TOTAL NUMBER OF TENDONS (1)	TOTAL NUMBER OF NEW TENDONS (2)	TOTAL NUMBER OF CONTROL TENDONS (3)
Circumferential	16	13	3
Top Cross Head	2	1	1
Bottom Cross Head	4	3	1
Longitudinal	15	12	3

NOTES:

1. The total number of tendons to be subjected to lift off testing shall be the indicated number in that group. With the exception of longitudinal tendons, all tendons designated for lift off testing shall have both end anchor assemblies lifted off if accessible. Longitudinal tendons will have only the top end anchor assembly lifted off.

2. The total number of new tendons consists of a tendon population selected at random for examination or testing over the next specified surveillance period. Selection shall be such that the total population of accessible tendons in that group shall be inspected/tested before beginning any repeat inspections/tests. These tendons may be included in the group of new tendons which are visually inspected, as indicated in Table 4.6.4-1.

3. The total number of control tendons consists of the same population of tendons in each tendon group that will be selected and will remain constant for all inspection/test surveillance cycles. The criteria for selection of these tendons shall be to select those tendons which represent conditions in which corrosion is most pronounced tempered by ready accessibility. These tendons will be included in the group of control tendons which are visually inspected, as indicated in Table 4.6.4-1. One of the tendons in the control group will be a tendon monitored by a load cell.

DRAFT
APR 1 1985

SURVEILLANCE REQUIREMENTS

4.6.4.2 The structural integrity of the PCRV shall be demonstrated by the following PCRV Tendon Load Cell surveillances:

a. At least once per 12 months, by functionally testing the load cell alarm circuit between the Data Acquisition System Room and the Control Room.

b. At least once per 5 years by checking for shift in load cell reference points for representative load cells.

c. Any abnormal degradation of the tendon load cells detected during the above required tests shall be reported to the Commission by submittal of a Special Report pursuant to Specification 6.9.2. This report shall include a description of the degradation, the inspection procedure and results, and the corrective actions taken.

4.6.4.3 The structural integrity of the PCRV shall be verified as follows:

a. Crack patterns on the visible surface of the PCRV shall be mapped prior to and following the initial proof test pressure (IPTP). Concrete cracks which exceed 0.015 inches in width shall be recorded. Subsequent concrete surface visual inspections shall be performed after the end of the first and third calendar year following initial power operation.

Recorded cracks shall be assessed for changes in length and any new cracks will be recorded. Additional inspections and mapping shall be conducted at ten calendar year intervals thereafter.

b. PCRV deformations and deflections at vessel midheight and at the center of the top head shall be monitored at five calendar year intervals during a vessel pressurization to operating pressure.

c. The PCRV support structure shall be visually examined for evidence of structural deterioration at ten calendar year intervals.

SURVEILLANCE REQUIREMENTS (continued)

4.6.4.4 The structural integrity of the PCRV liner shall be verified during the fifth refueling and every tenth refueling

DRAFT

APR 1 1985

thereafter by:

a. Removal and testing of three sets of twelve specimens with dosimeters from the approximately 750 specimens which have been placed adjacent to the outside surface of the top head liner for the purpose of determining irradiation-induced changes in notch toughness of the plate, weld metal, and heat affected zones.

b. The testing program shall meet the requirements of ASTM-E-185-82 to the extent practical for the configuration of the specimens and with the following exceptions:

1. Tensile specimens are not required.

2. Thermal control specimens are not required.

c. Any abnormal degradation of the liner detected during the testing shall be reported to the Commission within 30 days after the evaluation is completed by submittal of a Special Report pursuant to Specification 6.9.2. The Special Report shall contain a description of the test results, an evaluation of the effect of the degradation on the integrity of the liner, and a description of the corrective action to be taken, if any.

d. The interval for specimen removal and testing subsequent to the fifth refueling cycle may be adjusted based on the analysis of prior results.

4.6.4.5 PCRV integrity shall be demonstrated:

a. At least once per 12 months by:

1. Functionally testing and calibrating the instrumentation which monitors and alarms pressure in the core support floor and core support floor columns, AND

2. Functionally testing automatic isolation valves associated with pressurizing, purging, and venting PCRV penetration interspaces. However, this test may be performed at the next scheduled plant shutdown if these valves have not been tested during the previous year.

b. At least once per REFUELING outage by visually examining accessible portions of the refueling penetration hold down plate bolting.

DRAFT

APR 1 1985

c. At least once per 5 years by:

1. Surface examination (MT or PT) of accessible portions of the following welds in one steam generator penetration for indications of surface defects:

   a) The penetration shell to secondary closure weld

   b) The secondary closure to upper bellows support weld, and

   c) The lower bellows support to reheat header sleeve weld,

2. Visually examining accessible portions of the helium circulator restraint system (cylinder, ring, and bolting) for one penetration in each loop,

3. Testing the check valves on the HTFA purge lines,

4. Testing the check valves integral with HTFA or refueling penetrations, when such a penetration is open for refueling or maintenance if the check valves have not been tested in the last five years,

5. Visually examining the PCRV safety valve containment tank closure bolting for absence of surface defects when the tank is opened for testing of the PCRV overpressure protection assembly,

6. Determining the PCRV safety valve containment tank closure flange leak tightness following tank closure, and

7. Functionally testing the controls, position indication, and fail safe operation for remote manual isolation valves associated with pressurizing, purging, and venting PCRV penetration interspaces.

d. At least once per 10 years by:

1. Surface examination (MT or PT) of accessible portions of the following two welds in the bottom access penetration:

   a) The penetration shell to spherical head weld, and

DRAFT

APR 1 1985

b) The spherical head to closure flange weld,

2. Visually examining accessible portions of the bottom access penetration primary closure split ring assembly and its secondary closure bolting, AND

3. Visual examination of accessible portions of the PCRV safety valve penetration containment tank support components for indications of defects as follows:

   a) Surface examination (MT or PT) of the support skirt to tank attachment weld.

   b) Visual examination of the support skirt between the tank and PCRV outer wall.

   c) Visual examination, and torque or tension test the bolting attaching the support skirt to the PCRV outer wall.

DRAFT

APR 1 1985

e. At the end of the third and fifth years following initial power operation and every 10 years thereafter by examining the PCRV liner for corrosion induced thinning using ultrasonic inspection techniques.

f. At the end of the third year following inital power operation, and at 5 year intervals thereafter, by measuring the permeability of the PCRV concrete.

Surveillance Requirements 5.2.1.5.c.1, c.2, d.2, and d.3 shall be implemented per ISI Criterion C.

BASIS FOR SPECIFICATION LCO 3.6.4 / SR 4.6.4

Various monitoring programs are required to assure the structural integrity of the PCRV. These are specified in Section 5.13.4, 5.13.8 of Appendix E-17 of the FSAR. They consist primarily of monitoring and evaluating the PCRV with respect to tendon corrosion, tendon load cell surveillance, and concrete structure surveillance. Details of each program are identified in the appropriate Surveillance Requirement.

PCRV Tendon Corrosion

This surveillance requirement constitutes an interim accelerated inspection/testing program that has been put in place to monitor the prestressing system and develop a data base of tendon corrosion data.

Visual and lift off examinations of tendon assemblies provide assurance that the prestressing system has not degraded.

Tendon Load Cell

Since the relationship between effective prestress and PCRV internal pressure is directly related and easily calculated, monitoring tendon loads is a reliable means of assuring that the vessel is capable of containing its design pressure.

Monitoring tendon load changes will assure that corrosion, concrete strength reduction, or excessive steel relaxation has not occurred to the extent that they would compromise PCRV design. Load changes, as reflected by load cells, are monitored in the Control Room by an alarm system which alerts the operator if the allowable settings are exceeded. Upper settings will vary depending on the location of the tendon being monitored, while the lower settings for all load cells are set to correspond to 1.25 times the peak working pressure (PWP).

Concrete Integrity

Cracks are expected to occur in the PCRV concrete resulting from shrinkage, thermal gradients, and local tensile strains due to mechanical loadings. The degree of cracking expected is limited to superficial effects and is not considered detrimental to the structural integrity of the PCRV. Reinforcing steel is provided to control crack growth development with respect to size and spacing. Model testing has also shown that severely cracked vessels contain the normal working pressure for extended periods of time as long as the effective prestressing forces are maintained.

Cracks up to about 0.015 inches (limits of paragraph 1508b, ACI 318-63) for concrete not exposed to weather are generally considered acceptable and corrosion of rebars at such cracks is of negligible consequence. Large crack widths will require further assessment as to their significance, depending on the width, depth, length, and location of the crack on the structure, and must be considered with reference to the observed overall PCRV response.

Further discussion on the significance of concrete cracks in the PCRV is given in Section 5.12.5 of the FSAR.

Observed crack development with time during reactor operation will be related to the PCRV structural response as monitored by the installed sensors and deflection measurement. Details of the PCRV structural monitoring provisions are given in Section 5.13.4 and Appendix E-17 of the FSAR.

The interval for surveillance after the fifth year following initial prestressing may be adjusted based on the analysis of prior results.

Monitoring of overall PCRV deformations and deflection is the best indication of PCRV structural performance and verifies that the PCRV response is elastic and that no significant permanent strains exist.

Visual examination of the PCRV support structure will indicate that no structural deterioration has occurred. Significant cracking patterns or sizes should be investigated with respect to their impact on the integrity of the PCRV.

Liner Specimens

Irradiation experiments on liner material specimens indicated that the material was capable of fulfilling its function throughout the design life of the plant. Specimens with dosimeters have been placed adjacent to the outside surface of the top head liner to permit detection of any shifts of the NDT (nil ductility transition temperature) characteristics of liner plate, weld metal, and heat affected zone materials during the lifetime of the plant.

The testing program requires specimen removal and testing during the fifth refueling and at specified times thereafter which is adequate to detect significant changes. Tensile specimens are not required for ASTM-E-185-82 testing because the liner is not a load carrying member, but only a ductile membrane. Thermal control specimens are not required because the liner materials will normally be maintained at

or below 150 degrees F during all plant operations and there is no appreciable temperature cycling of the liner.

PCRV Integrity

The interval specified for functional test and calibration of the instrumentation and alarms monitoring the core support floor and columns will assure sending and alarming any change in their structural integrity.

The interval specified for valve testing is adequate to assure proper valve operation when isolation of the interspace auxiliary piping is required.

Structural integrity of PCRV penetration secondary pressure retaining boundaries is normally verified by continuous leakage monitoring and by periodic leakage testing of the penetration interspace. The specified examinations of accessible circumferential welds at structural discontinuities will provide additional assurance concerning the continued integrity of the secondary pressure boundary at these locations.

Examination of accessible penetration closures, flow restrictors, and equipment restraint or support components provides assurance that these components remain structurally sound and capable of performing their safety function under both normal and accident conditions.

Visual examination of the PCRV safety valve containment tank closure bolting and leakage examination of the tank closure flange provides assurance that containment tank integrity is restored after the tank cover has been reinstalled.

The ultrasonic inspection of the PCRV liner is provided to detect the thinning of the liner due to corrosion or to detect defects within the liner due to corrosion or to detect defects within the liner at representative areas. Although no corrosion is expected to occur, this specification allows for detection of corrosion or liner defects in the event of some unexpected and unpredicted changes in the liner characteristics. The provisions are discussed in Section 5.13 of the FSAR. The interval for surveillance after the fifth year following initial power operation may be adjusted based on the analysis of prior results.

Measurements of the relative helium permeability throughout plant life provides, as a supplement to other surveillance efforts, information concerning the continued integrity of the PCRV concrete. The interval for surveillance after the fifth year following initial power operation may be adjusted based on the analysis of prior results.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.5 REACTOR BUILDING CONFINEMENT

REACTOR BUILDING INTEGRITY

LIMITING CONDITION OF OPERATION

3.6.5.1 Reactor Building integrity shall be maintained.

APPLICABILITY: POWER OPERATION, LOW POWER, STARTUP, AND REFUELING

ACTION: Without Reactor Building integrity, restore Reactor Building integrity within 2 hours or be in SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Reactor Building integrity shall be demonstrated:

a. At least once per 31 days by;

1) Verifying that all Reactor Building louvers are in the closed position except as required by Specification 4.6.5.3.

2) Verifying that all external doors and hatches are in the closed position except as permitted by (b) below.

b. By verifying prior to each time that the truck doors to the truck bay are open that the reactor floor hatch, the deck hatch, and all personnel doors in the truck bay shall be closed.

DRAFT

APR 1 1985

BASIS FOR LCO 3.6.5.1 / SR 4.6.5.1

The integrity of the reactor building and operation of the ventilation exhaust system in combination limit the off-site doses under normal and abnormal conditions. In the unlikely event of a major release of activity from the PCRV, the combination of the reactor building integrity and ventilation exhaust system would act to keep off-site doses well below 10 CFR 100 limits (see FSAR Section 14.10.3.4).

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.5 REACTOR BUILDING CONFINEMENT

REACTOR BUILDING EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.2 The Reactor Building Exhaust system shall be OPERABLE with:

a. Reactor Building internal pressure subatmospheric, and

b. At least 2 exhaust fans (C-7301, C-7302, C-7302S) and the associated filter assemblies (F-7301, F-7302, F7302S) OPERABLE.

APPLICABILITY: At all times

ACTION: POWER OPERATION, LOW POWER, STARTUP

a. With Reactor Building internal pressure equal to or greater than atmospheric pressure, restore it to subatmospheric within 6 hours or be in at least SHUTDOWN within the next 24 hours.

b. With less than the required exhaust system equipment OPERABLE, restore the inoperable exhaust train(s) to OPERABLE within 7 days or be in at least SHUTDOWN within the next 24 hours.

SHUTDOWN AND REFUELING

With the Reactor Building internal pressure equal to or greater than atmospheric pressure or less than the required exhaust system equipment OPERABLE, immediately suspend CORE ALTERATIONS and handling of IRRADIATED FUEL in the Reactor Building.

SURVEILLANCE REQUIREMENTS

4.6.5.2 The Reactor Building exhaust system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the Reactor Building internal pressure is negative relative to atmospheric pressure.

DRAFT

APR 1 1985

b. At least once per 18 months all instruments necessary to monitor pressure in the Reactor Building shall be calibrated.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housing, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the exhaust system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than .05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the flow rate is 17,100 cfm plus or minus 10% per train.

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10%.

3. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Column while operating at a flow rate of 17,100 cfm plus or minus 10% for each filter train.

4. Verifying a flow rate of 17,100 cfm plus or minus 10% per train during system operation when tested in accordance with ANSI N510-1975.

d. After every 4400 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10%.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the exhaust system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol

DRAFT

APR 1 1985

while operating the system at a flow rate of 17,100 cfm plus or minus 10%.

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the exhaust system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 17,100 cfm plus or minus 10%.

g. After any structural modification to the filter housings, by verifying that the air flow distribution across the HEPA and charcoal filters is within plus or minus 20% when tested in accordance with ANSI N510-1975.

DRAFT

APR 1 1985

BASIS FOR LCO 3.6.5.2 / SR 4.6.5.2

The Reactor Building exhaust filter system is designed to filter the Reactor Building atmosphere prior to release to the facility vent stack during both normal and accident conditions of operation.

The building ventilation system maintains the Reactor Building pressure slightly subatmospheric and reduces the amount of radioactivity released to the environment, during normal operation or accident conditions.

DRAFT

APR 1 1985

PCRV AND CONFINEMENT SYSTEMS

3/4.6.5 REACTOR BUILDING CONFINEMENT

REACTOR BUILDING OVERPRESSURE PROTECTION SYSTEM

LIMITING CONDITION OF OPERATION

3.6.5.3 The Reactor Building overpressure protection system shall be OPERABLE with at least 70 of 94 louver panels OPERABLE:

APPLICABILITY: POWER OPERATION, LOW POWER AND STARTUP

ACTION: With the Reactor Building overpressure protection system inoperable, restore the system to OPERABLE status within 7 days, or be in SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3 The Reactor Building overpressure protection system shall be demonstrated OPERABLE:

a.) At least once per 92 days, by exercising each louver through one open-closed cycle.

b.) At least once per 18 months by simulating an overpressurization signal and verifying that each louver group opens fully at less than or equal to 3 inches of water gage differential pressure.

DRAFT

APR 1 1985

BASIS FOR LCO 3.6.5.3 / SR 4.6.5.3

The purpose of the pressure relief device is to maintain the integrity of the Reactor Building by relieving the pressure inside the building when it equals or exceeds 3 inches of water. In the unlikely event of the occurrence of a rapid increase of pressure inside the building of or exceeding 3 inches of water, the louvers would open, relieving the pressure, and then be automatically closed at approximately atmospheric pressure, restoring the integrity of the Reactor Building (see FSAR 6.2.3.4) and maintaining the potential doses from the occurrence to as low as practicable. The operability of 63 louvers is required to prevent the pressure buildup in the Reactor Building from exceeding design limits. Requiring at least 70 louvers to be operable provides a 10% margin of safety.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.1 TURBINE CYCLE

BOILER FEED PUMPS

LIMITING CONDITION FOR OPERATION

3.7.1.1 Two of the three boiler feed pumps shall be OPERABLE with at least one of the following combinations:

a. The motor driven boiler feed pump (P-3102) OPERABLE and one of the turbine driven boiler feed pumps (P-3101 or P-3103) OPERABLE, or

b. Two turbine driven boiler feed pumps (P-3101 and P-3103) OPERABLE and either auxiliary boiler OPERATING.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION: With both of the above combinations inoperable, restore either of the required combinations to OPERABLE status within 72 hours or be in at least STARTUP within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each boiler feed pump shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that each valve (manual, power-operated, or automatic) in the flow path to the emergency feedwater headers is in its correct position.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.1.1 / SR 4.7.1.1

Any one of the boiler feed pumps can supply feedwater for helium circulator motive power and steam generator heat removal to provide sufficient safe shutdown cooling. Emergency feedwater is required only in highly incredible depressurization accident (DBA-2) as discussed in FSAR Appendix D. Requiring a combination of two boiler feed pumps with backup steam supply when two steam driven pumps are used to meet this specification provides redundant capability for safe shutdown cooling.

Since the boiler feed pumps are operated during normal plant operations, a surveillance for checking valve positions once per 18 months is sufficient assurance of boiler feed pump availability. Other means of safe shutdown cooling are available using condensate or boosted firewater and physically redundant piping, valves and components. Operation in STARTUP results in low decay heat level, thus the redundant modes of safe shutdown cooling are sufficient for safe shutdown cooling in the event of any credible accident.

DRAFT

APR 1 1986

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.1 TURBINE CYCLE

STEAM/WATER DUMP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 The steam/water dump system shall be OPERABLE with the following:

a. A water level in the steam/water dump tank (T-2201) less than or equal to 45 inches;

b. At least one steam/water dump valve OPERABLE per loop (loop 1: HV-2215 or HV-2217, Loop 2: HV-2216 or HV-2218);

c. The steam/water dump tank safety valves (V-2270 and V-2275) at their proper setpoint (860 plus or minus 10 psig).

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION: With any of the above components of the steam/water dump system inoperable, restore the system to OPERABLE status within 72 hours or be in at least STARTUP within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The steam/water dump system shall be demonstrated OPERABLE:

a. At least once per 24 hours, by verifying that the tank level is less than or equal to 45 inches.

b. At least once per 18 months, by operating the steam/water dump valves through one complete cycle of full travel.

c. At least once per 5 years, by testing the steam/water dump tank safety valve setpoints.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.1.2 / SR 4.7.1.2

The steam/water dump system minimizes water in-leakage into the core as a result of a steam generator tube rupture (FSAR Section 6.4). Proper operation of the system minimizes damage to the core from the steam-graphite reaction and also reduces plant downtime required for subsequent cleanup of the primary coolant.

The condensate inventory in the dump tank cools the fluid dumped from the steam generator. There is no minimum level required since the final pressure after a dump into a dry vessel would not lift the dump tank safety valves to create a potential release path. A maximum level of 45 inches is established to prevent lifting the safety valves due to hydrostatically filling the tank during a dump.

The dump valves are required to open, permitting the steam generator to dump its contents into the dump tank. Only one valve is required to handle the inventory for each loop. Proper operation of the valves will minimize core damage and high primary system pressure in the event of a steam generator tube rupture.

The dump tank safety valves protect the integrity of the dump tank which may contain radioactive fluids. The steam/water dump tank pressure, temperature and radiation monitors are used to verify that the proper steam generator has been dumped in case of a steam generator tube rupture. It also prevents venting and draining of the tank to the radioactive gaseous and liquid systems before the contents have been adequately cooled. The radiation monitors (RIS-93250-12 and RIS-93251-12) are covered by Specification 4.3.2.2.

The superheater outlet stop check valves (HV-2223 and HV-2224), feedwater block valves (HV-2201 and HV-2202) and feedwater flow control valves (FV-2205 and FV-2206) limit the fluid inventory being dumped into the steam/water dump tank. The moisture monitors and pressure monitors initiate steam/water dump protective action and are covered by Specification 4.3.1.

DRAFT
APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.1 TURBINE CYCLE

PRESSURE RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.3 The main steam power operated relief valves (PV-22167 and PV-22168) and the hot reheat steam power operated relief valves (PCV-5221-1 and PCV-5221-2) shall be OPERABLE.

APPLICABILITY: POWER OPERATION

ACTION:

a. With either of the main steam power operated relief valves (PV-22167 or PV-22168) inoperable, restore the valve(s) to OPERABLE status within 72 hours or reduce power to less than 70% power.

b. With either hot reheat steam power operated relief valves (PCV-5221-1 or PCV-5221-2) inoperable, restore the valve(s) to OPERABLE status within 72 hours or be in at least STARTUP within the subsequent 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The main steam power operated pressure relief valves (PV-22167 and PV-22168) and the hot reheat steam power operated pressure relief valves (PCV-5221-1 and PCV-5221-2) shall be demonstrated OPERABLE at least once per 18 months by exercising each associated pilot valve (PC-22167, PC-22168, and pressure switch (PS-5221-1 and PS-5221-2), respectively.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.1.3 / SR 4.7.1.3

Overpressure protection for the main steam section of each steam generator loop is provided by ASME Code spring loaded safety valves. These safety valves will relieve approximately 110% of design pressure and 105% of normal loop flow. In addition, the main steam bypass system is designed to handle at least 75% of loop flow while discharging to the bypass flash tank (also protected by code safety relief valves). The bypass system will normally prevent either the mainsteam code safety relief valves or the power operated pressure relief valves from operating at loop flows/pressures equal to or less than 75%.

Isolation of the main steam power operated pressure relief valves during startup and through boilout is acceptable and desirable since the water/steam conditions through boilout (approx. 30%) can be quite damaging to the power operated pressure relief valves.

The hot reheat steam line is equipped with two power operated pressure relief valves actuated by turbine trip, high condenser pressure, high hot reheat steam pressure or loss of condensate pressure to the reheat desuperheaters. The hot reheat steam line is also equipped with six ASME Code spring loaded safety valves. These hot reheat steam relief valves discharge to atmosphere. Transients in which the hot reheat steam power operated relief valves are utilized are discussed in Sections 10.3.1 and 10.3.2 of the FSAR.

PLANT AND SAFE SHUTDOWN COOLING SUPPORT

DRAFT
APR 1 1985

3/4.7.1 TURBINE CYCLE

SECONDARY COOLANT ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The secondary coolant activity level shall be limited to 0.009 uCi/cc of DOSE EQUIVALENT I-131 and 6.8 uCi/cc of tritium.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION: With the specific activity of the secondary coolant system greater than 0.009 uCi/cc of DOSE EQUIVALENT I-131 or 6.8 uCi/cc of tritium, be in at least STARTUP MODE within 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 During operation in LOW POWER or POWER OPERATION, the specific activity of the secondary coolant system shall be determined to be within the above limits by analyzing for DOSE EQUIVALENT I-131, tritium, and gross beta plus gamma concentration:

a. At least once 7 days.

b. At least once per 24 hours if the secondary coolant activity level reaches 10% of the above limits. Weekly sampling may be resumed when the activity level decreases to less than 10% of the limits.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.1.4 / SR 4.7.1.4

The limitations on the secondary coolant activity have been established to ensure that the exclusion area boundary dose is less than the suggested limits in the event of an accident involving loss of outside power, main turbine trip, and failure of one diesel generator to start (FSAR Section 10.3.2). In that event, about 52,000 gallons of secondary coolant water would be vented to the atmosphere as steam. Assuming a dilution factor of 2.7 E-3, no partition factor of the iodine between the steam released and the water not released, a two hour exposure dose of about 1.5 Rem to the thyroid would be obtained. Using the same assumptions for tritium a two hour exposure dose of about 0.5 Rem to the whole body would be obtained.

The accident referenced above is initiated from a condition in which the turbine is operating, and therefore is applicable only to POWER OPERATION or LOW POWER.

A weekly surveillance interval is sufficient to monitor the activity of the secondary coolant when levels are below 10% of the limits of Specification 3.7.1.4. Adequately close monitoring of the secondary system is provided by the escalated (daily) sampling and analysis frequency if the activity level reaches 10% of the limits of Specification 3.7.1.4

Large leaks of activity into the secondary system would be detected and alarmed by instruments and systems as described in Section 7.3.5 of the FSAR.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.2 HYDRAULIC POWER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two hydraulic power systems, providing control to their respective coolant loops, shall be OPERABLE with:

a. One OPERABLE hydraulic valve accumulator and associated header servicing each group of valves,

b. Hydraulic fluid pressure to any group of valves maintained greater than 2500 psig, and

c. Two OPERABLE hydraulic pumps, and system fluid pressure maintained greater than 2500 psig.

APPLICABILITY: POWER OPERATION, LOW POWER and STARTUP

ACTION:

a. With no hydraulic valve accumulator or with loss of capability to supply at least 2500 psig to the valve operators of one group of valves, isolate the affected loop within 1 hour.

b. With less than two operable hydraulic pumps or inability to maintain the required hydraulic pressure, isolate the affected loop using manual and electrically operated valves within 1 hour.

SURVEILLANCE REQUIREMENTS

4.7.2 The hydraulic power system shall be demonstrated OPERABLE:

a. At least once per 92 days, by performing a functional test on the system pressure indicators and pressure alarms.

b. At least once per 18 months, by performing a CHANNEL CALIBRATION on the system pressure indicators and pressure alarms.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.2 / SR 4.7.2

Each secondary coolant loop hydraulic power system is designed with three hydraulic fluid pumps, two hydraulic accumulators for each group of hydraulic operated valves, and separate headers to each group of valves. The hydraulic system will normally operate with two hydraulic fluid pumps and both hydraulic accumulators in service. The third hydraulic pump and one accumulator in each group is redundant.

Loss of two hydraulic fluid pumps or both hydraulic accumulators servicing a group of valves indicates the potential for complete or partial loss of valve OPERABILITY in the affected secondary coolant loop. The one hour time interval prior to required affected loop isolation allows for an effort to regain OPERABILITY of a second hydraulic fluid pump and/or at least one accumulator for the affected valve group.

In the event hydraulic oil is lost to a group of valves, some degree of control will be lost and the affected secondary coolant loop is isolated. With only one group of valves inoperative, the ability to totally isolate the affected coolant loop is assured by the selective grouping of valves.

In the event of loss of all hydraulic power in one system, all flow pressure & speed control as well as ability to totally isolate the affected secondary coolant loop is lost. Therefore, the affected loop is isolated with the exception of cold reheat steam path to the condensor via the circulator steam drive bypass line. Heat removal is accomplished with the non-affected secondary coolant loop. Upon depletion of steam to drive the circulators, the circulator(s) are operated on their Pelton drives.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN SUPPORT SYSTEMS

3/4.7.3 INSTRUMENT AIR SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two instrument air systems, each consisting of one instrument air compressor, one instrument air receiver, one Reactor Building air header, and one Turbine Building air header, shall be OPERABLE.

APPLICABILITY: POWER OPERATION, LOW POWER and STARTUP

ACTION: With only one instrument air system OPERABLE, restore at least two systems to OPERABLE status within 72 hours or be in at least STARTUP within the next 12 hours and SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 Each instrument air system shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying that the instrument air system is pressurized to greater than or equal to 80 psig.

b. At least once per 18 months, by performing a system functional test which includes a simulated loss of header pressure, and verifying the automatic air compressor start and alarm functions.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.3 / SR 4.7.3

The OPERABILITY of the instrument air systems ensures that sufficient air is available for continued operation of essential instrumentation, control devices and pnuematic power operated valves required for safe shutdown cooling.

The instrument air system is a normally operating system. The system is functionally demonstrated OPERABLE by ensuring that system pressure is maintained at greater than 80 psig. A working pressure of greater than 80 psig ensures adequate air pressure for operating essential valves and instrumentation.

The monthly check of the system operating pressure coupled with the 18 month system functional test demonstrates that the system will behave as described in the safety analyses.

DRAFT
APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The service water system shall be OPERABLE with:

a. At least two of three service water pumps (P-4201, P-4202, P-4202S) OPERABLE, and

b. An OPERABLE flow path to SAFE SHUTDOWN COOLING users of service water (emergency diesel coolers, instrument air compressor and after coolers, and the PCRV liner cooling heat exchangers).

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION:

a. With only one service water pump OPERABLE, restore at least two pumps to OPERABLE status within 72 hours or be in at least STARTUP within the subsequent 24 hours.

b. Without an OPERABLE flow path to SAFE SHUTDOWN COOLING users of service water, restore an OPERABLE flow path within 72 hours or be in at least STARTUP within the subsequent 24 hours.

SURVEILLANCE REQUIREMENTS

No specific surveillances are required as the service water system is normally operating, including the required OPERABLE flow paths. The flow path to the standby diesel generators is demonstrated OPERABLE by the surveillance testing of the diesel generators (Specification 4.8.1.1.2).

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.4

About 25% capacity of a single service water pump is sufficient to perform SAFE SHUTDOWN COOLING. The SAFE SHUTDOWN COOLING users of service water are the emergency diesel coolers, the instrument air compressors and after coolers and the PCRV liner cooling heat exchangers. The instrument air compressors and after coolers and the PCRV liner cooling heat exchangers are continuously supplied with service water during normal plant operation. Service water to the emergency diesel coolers is verified by surveillance testing of the emergency diesels. With the plant at LOW POWER or POWER OPERATION, there will normally be at least two service water pumps in OPERATION. The requirements for two OPERABLE service water pumps is for single failure considerations. Circulating and firewater systems serve as backups to the service water system. SAFE SHUTDOWN COOLING, including service water requirements, is discussed in FSAR Section 10.3.9. The service water system is discussed in FSAR Section 9.8. The service water system is not covered during STARTUP because of system redundancies. The firewater system provides an independent source of cooling water for all safe shutdown essential water requirements. The firewater system is applicable during all modes of operation.

DRAFT
APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.5 PRIMARY COOLANT DEPRESSURIZATION

LIMITING CONDITION FOR OPERATION

3.7.5 a. An OPERABLE flow path for primary coolant depressurization shall exist from the primary coolant system through at least one helium purification train to the Reactor Building Ventilation System exhaust, and

b. At least 650 gallons of liquid nitrogen shall be maintained in the Liquid Nitrogen Storage Tank (T-2501)

APPLICABILITY: POWER OPERATION, LOW POWER, and STARTUP

ACTION:

a. With both helium purification trains inoperable, restore at least one train to OPERABLE status or be in at least STARTUP within 12 hours and SHUTDOWN within the next 12 hours.

b. With less than 650 gallons of liquid nitrogen in the Nitrogen Storage Tank, restore the liquid nitrogen storage inventory to 650 gallons or be in at least STARTUP within 12 hours and SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 OPERABILITY of the Helium Purification System shall be verified:

a. At least once per 24 hours by determining that the contents of the Liquid Nitrogen Storage Tank (T-2501) are at least 650 gallons.

b. At least once per 18 months by cycling (through one complete cycle of full travel) valves for routing helium gas to the Reactor Building Ventilation exhaust and for cooling the High Temperature Filter Adsorber (HTFA).

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.5 / SR 4.7.5

The Helium Purification System is a normally operating system with redundant backups and requires no tests or inspections beyond good power plant operating and maintenance practices to verify operability (FSAR Section 9.4.8).

In the unlikely event of an accident involving the extended loss of forced circulation, the Helium Purification System must be OPERABLE to accomplish depressurization of the primary coolant system. The normal depressurization flow path includes the High Temperature Filter Adsorber, the Helium Purification Cooler, the Helium Purification Dryer, the Low Temperature Gas-to-Gas Heat Exchanger, the Low Temperature Adsorber, and associated piping and valves leading to the reactor building exhaust. If both purification trains are inoperable other compensatory measures (such as reducing the buffer supply to operating circulators) may be taken to minimize the increase in PCRV pressure. This is acceptable for a limited period of time due to the availability of an alternate depressurization flow path via the regeneration piping.

A total of 650 gallons of liquid nitrogen is required to provide refrigeration for the Low Temperature Adsorber during depressurization without nitrogen recondensing capability.

The only aspect of system operation that must be monitored is the maintenance of the required quantity of liquid nitrogen in the liquid nitrogen storage tank and the operability of isolation valves for routing helium gas to the reactor building ventilation exhaust and for cooling the HTFA.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.6 FIRE SUPPRESSION SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.1 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. Dry Pipe Systems (automatically actuated valves)

1. The 4160/480 volt AC transformers.

2. The reactor plant exhaust filters.

3. The turbine lube oil reservoir room.

4. The turbine lube oil storage room.

5. The reserve auxiliary transformer.

6. The main/unit auxiliary transformer.

b. Wet Pipe Systems (fused spray heads)

1. Hydraulic power unit 1A.

2. Hydraulic power unit 1B.

3. Steam driven boiler feed pump 1A.

4. Steam driven boiler feed pump 1C.

5. The auxiliary boiler room.

6. The hydrogen seal oil unit.

7. The helium circulator turntable reservoir.

c. Manually Operated Fixed Spray Systems

1. The 480 volt switchgear room.

2. The turbine building side of the "G" wall for the congested electrical cable area.

3. The reactor building side of the "J" wall for the congested electrical cable area.

DRAFT
APR 1 1985

4. The auxiliary electrical equipment room.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour:

1. Establish a continuous fire watch with backup fire protection equipment, or

2. Perform an evaluation to determine that the area protected by the inoperable Spray/Sprinkler System contains no safety related equipment which is required to be OPERABLE, in which case no action is required, or

3. Perform an evaluation to determine that the area protected by the inoperable Spray/Sprinkler System includes no redundant systems or components, and establish an hourly fire watch patrol.

b. The above evaluations may be performed subsequent to the establishment of a continuous fire watch. Where reduced actions can be justified, they may be initiated at that time.

c. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each of the required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position, and

b. At least once per 12 months, by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

a. Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated actuation signal, and

DRAFT

b. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel. 1985

2. By visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years, by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.6.1 / SR 4.7.6.1

The OPERABILITY of the Spray and Sprinkler System ensures that adequate fire suppression cabability is provided by wet pipe sprinklers and dry pipe spray nozzles in areas of the plant where this system protects safety-related equipment, or equipment that can perform a safety function.

The Spray and Sprinkler System will minimize potential damage to safety related equipment or equipment that can perform a safety function should a fire occur.

If one or more of the systems becomes inoperable, a continuous fire watch with backup fire suppression equipment will be provided for the inoperable system.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.6 FIRE SUPPRESSION SYSTEMS

CARBON DIOXIDE SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.2 The Emergency Diesel Generator Rooms' Carbon Dioxide system shall be OPERABLE.

APPLICABILITY: Whenever the emergency diesel generators are required to be OPERABLE.

ACTION:

a. With the above required carbon dioxide system inoperable, within 1 hour establish an hourly fire watch with backup fire suppression equipment for the affected Emergency Diesel Generator Room(s).

b. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.2 The Emergency Diesel Generator Rooms' carbon dioxide fire suppression system shall be demonstrated OPERABLE:

a. At least once per 7 days, by verifying the carbon dioxide storage tank level to be greater than 30% and pressure to be greater than 285 psig, and

b. At least once per 18 months, by verifying

1. The system, including valves and associated ventilation system fire dampers, actuates manually and automatically upon receipt of a simulated actuation signal, and

2. Flow from each nozzle during a "Puff Test."

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.6.2 / SR 4.7.6.2

The OPERABILITY of the carbon dioxide suppression system ensures that postulated fires in either of the emergency diesel generator rooms can be automatically suppressed. In the event that the carbon dioxide system becomes inoperable, an hourly fire watch coupled with backup or portable fire fighting equipment will provide an equivalent level of fire protection for postulated fires.

A weekly check of the level and pressure in the carbon dioxide storage tank ensures that an adequate quantity of carbon dioxide is available to suppress any postulated fires in the emergency diesel generator rooms. Periodic performance of a "puff test" ensures that the distribution headers are not blocked. Periodic verification that the system reacts to a simulated actuation signal provides adequate assurance that carbon dioxide concentration of 34% by volume will be maintained.

DRAFT
APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.6 FIRE SUPPRESSION SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.3 The following Halon systems and associated HVAC Isolation Dampers shall be OPERABLE:

a. Control Room

b. Auxiliary Electric Room

c. 480 Volt Switchgear Room

d. Building 10 - Switchgear Room and Ground Level

e. Building 10 - Ground Level Under Mezzazine Floor

f. Building 10 - Battery Room.

APPLICABILITY: At all times

ACTION:

a. With one or more of the above required Halon systems or HVAC Isolation Dampers inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for the affected room(s).

b. The provisions of Specification 3.0.5 and 3.0.6 are not applicable.

DRAFT

APR 1 1983

SURVEILLANCE REQUIREMENTS

4.7.6.3 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 184 days by verifying Halon storage tank weight to be at least 95% full charge weight (or level) and pressure to be at least 90% of full charge pressure.

b. At least once per 18 months, by:

1. Verifying that the system, including associated HVAC Isolation Dampers, actuates correctly, upon receipt of a simulated test signal, and

2. Verifying that the distribution headers and nozzles are not blocked by flowing air through the system.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.6.3 / SR 4.7.6.3

The OPERABILITY of the Halon systems ensures that adequate fire suppression capability is available for all postulated fires in the Three Room Control Complex and in Building 10. The Halon systems consists of two main distribution systems, one for Building 10 and the other for the Three Room Control Complex. The main distribution systems are separated into independent subsections that provide Halon to the following fire areas that house safety-related equipment; 1) the Control Room, 2) the Auxiliary Electric Room, 3) the 480V Switchgear Room, 4) Building 10's Switchgear Room and Ground Level, 5) Building 10's Ground Level under the Mezzazine Floor, and 6) Building 10's Battery Room. The OPERABILITY of the associated HVAC isolation dampers ensure that adequate room isolation will be available to maintain an effective concentration of Halon after actuation of the suppression system. In the event that portions of the Halon suppression systems are inoperable, backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. An installed sprinkler system (Specification 3.7.6.1) provides dedicated backup suppression for the 480 volt Switchgear Room and the Auxiliary Electric Room.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Halon suppression system are met. An allowance is made for ensuring a sufficient volume of Halon is in the storage tanks by verifying either the weight or the level of the tanks. Verification that the distribution headers are not blocked demonstrates their ability to spray Halon when needed to suppress a fire. Verification that the system, and its associated HVAC isolation dampers react to a simulated actuation signal will ensure overall system response to a postulated fire.

DRAFT

APR 1 1986

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.6 FIRE SUPPRESSION SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.6.4 The fire hose stations given in Table 3.7.6-1 shall be OPERABLE.

APPLICABILITY: At all times

ACTION: With any of the fire hose stations listed in Table 3.7.6-1 inoperable:

a. Ensure that any affected safety-related equipment can be protected from an adjacent hose station, or

b. Route additional equivalent capacity hose from the nearest hose station to protect the affected safe shutdown equipment within one hour.

c. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.4 Each of the fire hose stations required above shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations to assure all required equipment is at the stations.

b. At least once per 18 months, by:

1. Removing the hose for inspection and re-racking, and

2. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

2. Conducting a hose hydrostatic test at a pressure of 175 psig.

DRAFT

APR 1 1985

TABLE 3.7.6-1

FIRE HOSE STATIONS

HOSE STATION NO.	BUILDING	ELEVATION
TH7 E2	Turbine	4829
TH7 F6	Turbine	4829
TH7 B6	Turbine	4829
TH7 C2	Turbine	4829
TH6 C2	Turbine	4811
TH6 E2	Turbine	4811
TH6 G6	Turbine	4811
TH6 B6	Turbine	4811
TH5 E2	Turbine	4791
TH5 G3	Turbine	4791
TH5 G6	Turbine	4791
TH5 B6	Turbine	4791
TH5 C2	Turbine	4791
TH12 G4	Access Bay	4885
TH11 G3	Access Bay	4885
TH10 H4	Access Bay	4864
TH10 G3	Access Bay	4864
TH8 G4	Access Bay	4846
TH11 G6	Access Bay	4885
TH14 G3	Access Bay	4940
TH15 J4	Access Bay	4960
RH13 M2	Reactor	4916
RH13 J2	Reactor	4916
RH12 J2	Reactor	4906

DRAFT

APR 1 1985

TABLE 3.7.6-1 (continued)

HOSE STATION NO.	BUILDING	ELEVATION
RH11 M5	Reactor	4881
RH11 M2	Reactor	4881
RH11 K2	Reactor	4881
RH11 J5	Reactor	4881
RH10 M5	Reactor	4864
RH10 M3	Reactor	4864
RH10 K2	Reactor	4864
RH10 J5	Reactor	4864
RH9 M5	Reactor	4854
RH9 J5	Reactor	4849
RH8 M5	Reactor	4839
RH8 J5	Reactor	4839
RH7 M5	Reactor	4829
RH7 M3	Reactor	4829
RH7 J2	Reactor	4829
RH7 J5	Reactor	4829
RH6 M3	Reactor	4811
RH6 J2	Reactor	4811
RH5.5 M3	Reactor	4801
RH5.5 J2	Reactor	4801
RH5 M5	Reactor	4791
RH5 M3	Reactor	4791
RH5 J5	Reactor	4791

DRAFT

APR 1 1985

TABLE 3.7.6-1 (continued)

HOSE STATION NO.	BUILDING	ELEVATION
RH4 L5	Reactor	4781
RH4 J5	Reactor	4781
RH3 M5	Reactor	4771
RH3 J5	Reactor	4771
RH3 M3	Reactor	4769
RH3 J2	Reactor	4769
RH2 M3	Reactor	4759
RH2 J2	Reactor	4759
RH2 M5	Reactor	4756
RH2 J5	Reactor	4756
RH1 M5	Reactor	4740
RH1 M3	Reactor	4740
RH1 K3	Reactor	4740
RH1 J5	Reactor	4740

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.6.4 / SR 4.7.6.4

The OPERABILITY of the Fire Hose Stations ensures that adequate fire suppression capability is provided throughout the turbine and reactor buildings.

The Fire Hose Stations will minimize potential damage to safety-related equipment should a fire occur.

If one or more of the Fire Hose Stations becomes inoperable, backup fire suppression coverage will be provided by an adjacent OPERABLE hose station.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.6 FIRE SUPPRESSION SYSTEMS

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.6.5 The following yard fire hydrants and associated hydrant hose houses shall be OPERABLE:

a. Number 1 Northeast of the circulating water valve pit.

b. Number 3 North of the circulating water cooling tower.

c. Number 6 South of the circulating water cooling tower.

d. Number 7 Southeast of the service water cooling tower.

e. Number 11 Southeast of the turbine building.

APPLICABILITY: At all times

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses listed above inoperable, within one hour have sufficient additional lengths of hose located in an OPERABLE hydrant hose house to provide service to the unprotected area(s).

b. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.5 Each of the above yard fire hydrants and associated hydrant hose houses listed above shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the nydrant hose house to assure all required equipment is at the hose house,

DRAFT

APR 1 1985

b. At least once per 184 days, by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged, and

c. At least once per 12 months, by:

1. Conducting a hose hydrostatic test at a pressure of at least 75 psig,

2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings, and

3. Performing a flow check of each hydrant to verify its OPERABILITY.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.6.5 / SR 4.7.6.5

The OPERABLITY of the Yard Fire Hydrants and Hydrant Hose Houses ensures that adequate fire suppression capability is provided throughout the yard area.

If one or more of the Yard Fire Hydrants or associated Hydrant Hose Houses becomes inoperable, backup fire suppression coverage will be provided by an adjacent OPERABLE Hydrant.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.7 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.7 All fire barriers (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant safe shutdown systems within a fire area and all sealing devices in barrier penetrations (fire doors, fire dampers, cable, piping, and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times

ACTION:

a. With one or more of the above required fire barriers and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected barrier, or verify the OPERABILITY of fire detectors on at least one side of the inoperable barrier and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7 The above required fire barriers and penetration sealing devices shall be determined OPERABLE:

a. At least once per 24 hours, by verifying the position of each unlocked closed fire door.

b. At least once per 7 days, by verifying the position of each locked fire door.

c. At least once per 18 months, by performing a visual inspection of the exposed surfaces of:

1. Each fire barrier,

2. Each fire damper and associated hardware, and

DRAFT

APR 1 1985

3. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection on an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration will be inspected every 15 years.

d. Following any maintenance or repair work which disturbs the fire retardant material in the sealed penetrations, by verifying that the seal is returned to its previous condition.

BASIS FOR SPECIFICATION LCO 3.7.7 / SR 4.7.7

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited, as analyzed in the Fire Hazard Analysis. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment.

In the event that a fire barrier does not remain intact, a continuous fire watch on one side of the affected barrier or an hourly fire watch patrol in conjunction with OPERABLE fire detectors will ensure early notification of a potential fire hazard.

The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABLITY.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.8 ACM DIESEL GENERATOR

LIMITING CONDITION FOR OPERATION

3.7.8 The ACM diesel generator unit shall be OPERABLE with:

a. A flow path from a fuel oil storage tank through an OPERABLE fuel oil transfer pump to the diesel fuel oil day tank, and

b. A minimum fuel oil inventory in the fuel oil storage tanks of at least 10,000 gallons.

APPLICABILITY: POWER OPERATION, LOW POWER AND STARTUP

ACTION:

a. With the above ACM diesel generator requirement not satisfied, within 1 hour establish a fire watch patrol to inspect the CONGESTED CABLE AREAS at least once per hour, and

b. The provisions of Specification 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8 The ACM diesel generator unit shall be demonstrated OPERABLE:

a. At least once per 31 days, by:

1. Verifying the contained fuel volume in the fuel oil storage tanks.

2. Verifying the fuel oil transfer pump starts and transfers fuel from the storage system to the diesel fuel oil day tank.

3. Verifying the diesel starts from ambient conditions, idles and accelerates to an engine speed of at least 900 rpm. The generator voltage and frequency shall be 4160 plus or minus 416 volts and 60 plus or minus 1.2 Hz.

DRAFT

APR 1 1985

4. Verifying the generator is synchronized, loaded to greater than 675 KW, and operates at load for an additional 60 minutes.

b. At least once per 92 days, by obtaining a sample of fuel oil in accordance with ASTM-D270-1975, and by verifying that the sample meets the following minimum requirements:

1. The sample, when tested in accordance with the test specified in ASTM-D975-1977, has a water and sediment content of less than or equal to 0.05 volume percent and a kinematic viscosity at 40 degrees C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes.

2. The sample, when tested in accordance with ASTM-D2274-1970, has an impurity level of less than 2 mg of insolubles per 100 ml.

c. At least once per 18 months, by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

2. Verifying that each of the required ACM loads listed on Table 4.7.8-1 can be energized via the ACM electrical distribution system.

d. At least once per 10 years, by emptying the fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or equivalent.

DRAFT
APR 1 1985

TABLE 4.7.8-1

ALTERNATE COOLING METHOD - DIESEL GENERATOR LOADS

a.	Fire Water Pump	(P-4501)
b.	Service Water Pump	(P-4201 or P-4202)
c.	Service Water Tower Fan	(C-4201X or C-4202X)
d.	Service Water Return Pump	(P-4203 or P-4204)
e.	PCRV Liner Cooling System Pumps (2)	(P-4601 or P-4601S) and (P-4602 or P-4602S)
f.	Circulating Water Makeup Pump	(P-4118 or P-4118S)
g.	Reactor Plant Exhaust Fan	(C-7301 or C-7302)
h.	Diesel Oil Transfer Pumps (2)	(P-4803 or P-4804)
i.	Helium Purification Cooling Water Pump	(P-4701 or P-4702)
j.	Firewater Pump House Vent Fans (2) & Louvers	(C-7521 and C-7522)
k.	Motor Operated Valve	(HV-2301 or HV-2302)
l.	Stack Effluent Radiation Monitor	(PING-1)
m.	ACM Plant Lighting	
n.	Breathing Air Compressors	(C-4501P or C-4502P)
o.	Start-up Battery for Diesel Generator and D.C. control.	

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.8/ SR 4.7.8

The OPERABILITY of the ACM diesel generator ensures that a reliable and independent source of power will be available to reactor core cooling equipment and instrumentation in the event of an incapacitating fire in the THREE-ROOM CONTROL COMPLEX or other congested electrical areas. The ACM diesel generator provides power independent of the normal plant electrical distribution system by manually repositioning electrical transfer switches.

In the event that the ACM diesel generator is inoperable, a fire watch is established. The inspection of CONGESTED CABLE AREAS by a fire watch patrol ensures that a major fire will not develop in these areas by the ability to immediately detect and suppress fires before they can incapacitate the normal plant electrical distribution system. The Halon System (Specification 3/4.7.6.3) and the Spray and/or Sprinkler Systems (Specification 3/4.7.6.1) provide adequate suppression for the CONGESTED CABLE AREAS.

Ensuring that a fuel oil transfer system is OPERABLE with 10,000 gallons of fuel provides for over 72 hours of diesel generator operation with full ACM load. This is considered more than adequate time for obtaining additional fuel from off-site sources.

The Surveillance Requirements are adequate for demonstrating the OPERABILITY of the ACM diesel generator to perform its intended function. The testing of the diesel generator unit simulates, where practical, the parameters of operation and the physical environments that would be expected if an actual demand was to be placed on the system. Monthly testing of the diesel generator unit during normal plant operation, demonstrates proper startup and load-carrying cabability, and verifies that the required voltage and frequency are attained. This test also verifies that the components of the diesel generator unit are operable. A "diesel generator unit" consists of the engine, generator, combustion air system, cooling system, fuel supply system, lubricating oil system, starting battery system, automatic and manual controls, and the diesel generator breaker. Periodic energization of the ACM loads ensures that power can be distributed from the diesel to the ACM components upon demand. The explicit requirements on the fuel oil system provide adequate assurance that the diesel generator unit will continue to supply reliable electric power for the duration of the ACM event.

DRAFT

APR 1 1995

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.9 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 The control room ventilation system shall be OPERABLE in the minimum makeup mode with:

a. The control room emergency filter fan (C-7506) or the control room supply fan (C-7504X), and

b. The control room makeup ventilation filters (F-7502).

APPLICABILITY: At all times

ACTION: STARTUP, LOW POWER and POWER OPERATION

a. With the above requirements for the control room ventilation system not met, restore the system to OPERABLE status within 7 days or be in SHUTDOWN within the next 24 hours.

ACTION: SHUTDOWN and REFUELING

a. With the above requirements for the control room emergency ventilation system not met, restore the system to OPERABLE status within 7 days or suspend CORE ALTERATIONS, positive reactivity changes or handling of IRRADIATED FUEL.

b. The provisions of Specification 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9 The control room ventilation system shall be demonstrated OPERABLE as follows:

a. At least once per 18 months, or after any structural maintenance on the HEPA filter or charcoal adsorber housings, or following painting, fire, or a chemical release through the system by:

DRAFT

APR 1 1985

1. Verifying that the ventilation system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05 percent and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 500 ACFM plus or minus 10 percent.

2. Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodine penetration of less than 5 percent at 30 degrees C, 95% RH.

3. Verifying that the pressure drop across the HEPA filters and charcoal adsorbers is less than six (6) inches of water while operating the system at a flow rate of 500 ACFM plus or minus 10 percent.

4. Verifying a system flow rate of 500 ACFM plus or minus 10 percent during system operation when tested in accordance with ANSI N510-1980.

b. After each complete or partial replacement of a HEPA filter bank verify that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05 percent in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 500 ACFM plus or minus 10 percent.

c. After each complete or partial replacement of a charcoal adsorber bank verify that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05 percent in accordance with ANSI N510-1980 for a hyogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 500 ACFM plus or minus 10 percent.

DRAFT

APR 1 1985

d. At least once per 18 months, by verifying that the ventilation system maintains the control room at a positive pressure of greater than or equal to 0.02 inches water gauge with:

1. Only the control room supply fan in service.

2. Only the emergency filter fan in service.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.9 / SR 4.7.9

The control room ventilation system is designed to supply filtered, recirculated air at a positive pressure with minimum makeup. Makeup air is taken from the turbine building and passed through a prefilter, a high efficiency particulate air (HEPA) filter, and a charcoal adsorber thereby ensuring that control room personnel airborne radiation exposures during and following all credible accident conditions will not exceed 10 CFR 20 limits.

The control room pressure can be maintained positive with either the control room supply fan, or the emergency filter fan. Other design features permit isolation or recirculation of the control room ventilation system to maintain control room habitability. These modes of operation can be used if various failures of the control room emergency ventilation system components occur (See FSAR Section 11.2.2 and Appendix C, Criterion 11).

Assuring that an excessive pressure drop does not exist across the emergency makeup filter demonstrates that the filters and adsorbers are not clogged and that the control room pressure can be maintained positive. The specified surveillance tests are adequate to ensure system OPERABILITY under normal and abnormal conditions. Normal makeup requirements to maintain a positive pressure in the control room have been calculated to be 450 ACFM.

The addition of these surveillance requirements is in response to NUREG-0737, Item III.D.3.4.

DRAFT

APR 1 1985

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 All safety related snubbers (Class I and Ia) shall be OPERABLE.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION: With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.10.g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each safety related snubber shall be demonstrated OPERABLE by performance of the following inservice inspection program.

a. Snubber Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

The first inservice visual inspection of all snubbers shall be performed within six months from issuance of this Technical Specification. For the purpose of entering the schedule described in this section it shall be assumed that the facility has been on a six month inspection interval. If less than two subbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed at the next refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

DRAFT

APR 1 1985

No. of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period* #
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional.

Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.10.f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

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* The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

The provisions of Specification 4.0.4 are not applicable.

DRAFT

APR 1 1985

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

At least once per 18 months a representative sample of snubbers of each type shall be tested using one of the following sample plans.* The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period, or the sample plan used in the prior test period shall be implemented.

1. At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.10.f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

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* These sample plans shall supersede the previous requirements within 22 1/2 months of the issuance of this Technical Specification; the previous requirements shall continue to be implemented until superseded.

DRAFT

APR 1 1985

2. A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.10-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.10.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.10-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region or all the snubbers of that type have been tested. Should testing equipment failure invalidate functional testing, testing can resume anew at a later time provided all snubbers tested with the failed equipment are retested.

   The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test, but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

   The snubber functional test shall verify that:

   1. Activation (restraining action) is achieved within the specified range in both tension and compression;

   2. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

DRAFT

APR 1 1985

3. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers, irrespective of type, which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.10.e. for snubbers not meeting functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

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APR 1 1985

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.3.

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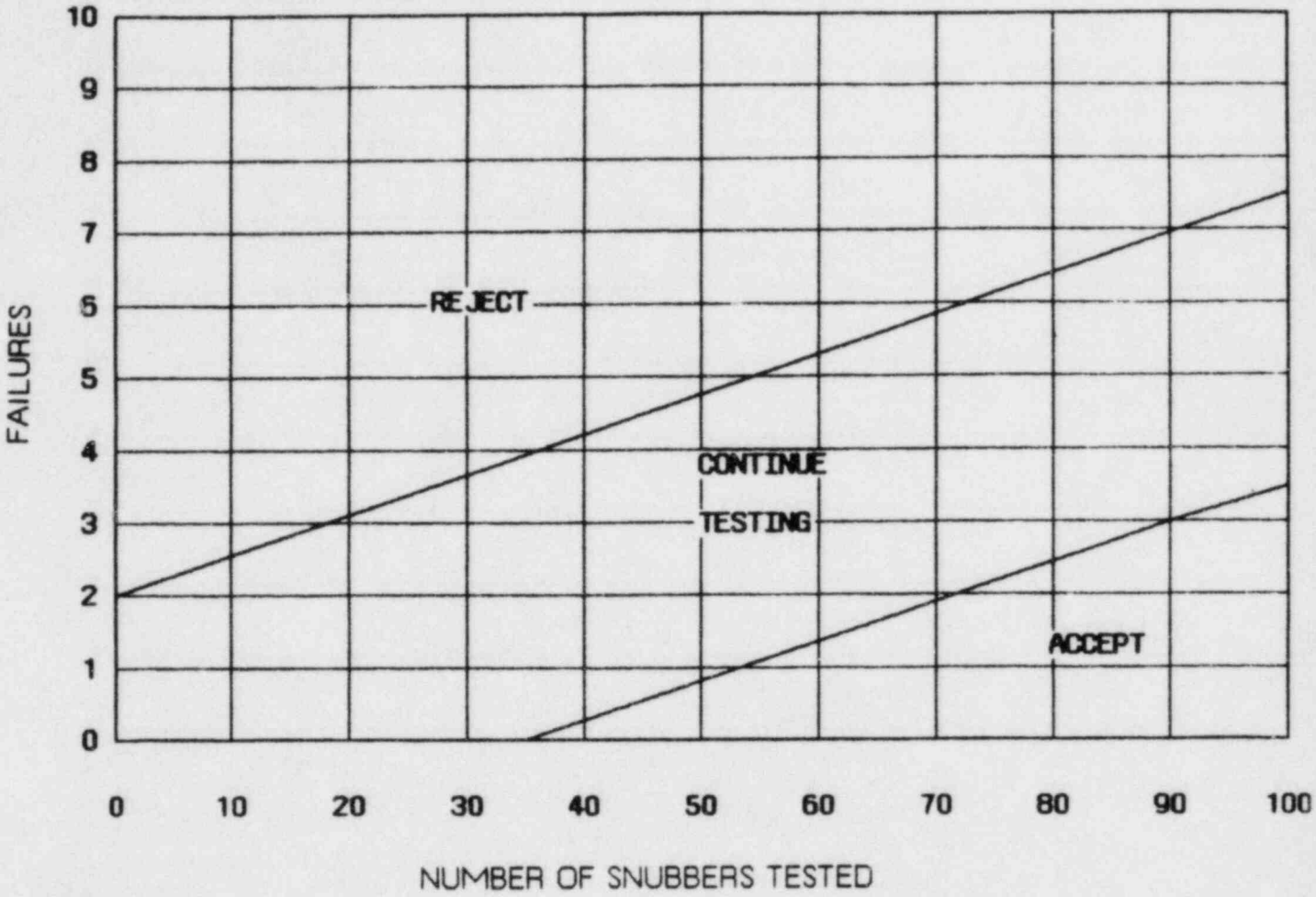
APR 1 1985



SAMPLE PLAN FOR NUMBER OF SNUBBERS TESTED

Figure 4.7.10-1

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APR 1 1985

BASIS FOR SPECIFICATION LCO 3.7.10 / SR 4.7.10

All safety related snubbers are required OPERABLE to ensure that the structural integrity of Class I systems is maintained during and following a seismic or other event intiating dynamic loads.

The consequence of an inoperable snubber is an increase in the probability of structural damage to piping resulting from the dynamic loads. It is, therefore, necessary that all snubbers required to protect the Class I systems, subsystems, or components be OPERABLE during reactor POWER OPERATION.

Because snubber protection is required only during relatively low probability events, a period of 72 hours is allowed for repair or replacement.

All Class I and Class Ia hydraulic and mechanical snubbers are visually inspected for overall integrity and OPERABILITY. The inspection includes verification of proper orientation, adequate hydraulic fluid level, when applicable, and proper attachment of snubber to piping and structures. Class I snubbers directly support Class I systems. Class Ia snubbers are those snubbers supporting non-Class I systems whose failure could adversely effect Class I systems.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next inspection. However, the results of an early inspection performed before the original required time interval has elapsed (nominal time less 25 percent) may not be used to lengthen the required inspection interval. Any inspection where results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability one of two functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

DRAFT

APR 1 1985

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7.10-1.

Figure 4.7.10-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in a list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

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AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.1 A.C. POWER SOURCES

A.C. POWER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a miminum, the following A.C. electrical power sources shall be OPERABLE:

a. The Unit Auxiliary Transformer* and the Reserve Auxiliary Transformer, and

b. Two separate and independent diesel generator-sets with:

1. Each diesel fuel oil day tank containing a minimum volume of 325 gallons of fuel,

2. A minimum of 20,000 gallons of diesel fuel in underground storage and an OPERABLE flow path(s) capable of transferring the fuel oil from storage to each day tank, and

3. An OPERABLE water-jacket heater for each diesel engine.

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION:

a. With either the Unit Auxiliary Transformer* the Reserve Auxiliary Transformer or a diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.1a. and by verifying that the diesel generator(s) is in auto-start mode within 1 hour and at least once per 8 hours thereafter; restore at least two A.C. electrical power sources and two diesel generators to OPERABLE status within 72 hours or be in at least STARTUP within the following 12 hours.

DRAFT

APR 1 1985

b. With either the Unit Auxiliary Transformer* or the Reserve Auxiliary Transformer and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.1a. and by verifying that the remaining diesel generator is in auto-start mode within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least STARTUP within the next 24 hours. Restore at least two A.C. electrical power sources and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least STARTUP within the following 12 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that within 2 hours all the required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, or be in at least STARTUP within the next 12 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. electrical sources by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 6 hours or be at least STARTUP within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 The Unit Auxiliary Transformer and the Reserve Auxiliary Transformer shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying correct breaker alignments, indicated power availability, and

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* The UAT may be out of service for up to 12 hours for the purpose of performing switching operations without being declared inoperable.

DRAFT

APR 1 1985

b. At least once per 18 months by verifying automatic transfer of house power supply from the Unit Auxiliary Transformer to the Reserve Auxiliary Transformer, verifying that the Unit Auxiliary Transformer's generator links can be removed.

4.8.1.1.2 Each diesel generator-set shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the diesel oil day tanks,

2. Verifying the fuel oil quantity in storage,

3. Verifying the capability to transfer fuel oil from the storage system to the diesel oil day tank,

4. Verifying the water-jacket heaters are OPERABLE by ensuring that the coolant water is being maintained at a temperature of greater than or equal to 120 degrees F,

5. Verifying the diesel engines start from a normal pre-heated condition and accelerate to normal operating speed. The generator voltage and frequency shall be 480 plus or minus 48 volts and 60 plus or minus 1.2 Hz. within 10 seconds after the start signal,

6. Verifying the OPERABILITY of the diesel engine's "shutdown" and "declutch" protective functions,

7. Verifying the generator is synchronized, loaded to greater than or equal to 1200 KW with two diesels per generator and operates for at least 60 minutes.

b. At least once per 92 days by verifying that a sample of fuel oil obtained in accordance with ASTM-D270-1975 has a water and sediment content of less than or equal to .05 volume percent and a kinematic viscosity @ 40|mdeg|xC of greater than or equal to 1.9 but less than or equal to 4.1 when tested in accordance with ASTM-D975-1977, and an impurity level of less than 2 mg. of insolubles per 100 ml. when tested in accordance with ASTM-D2274-1970.

c. At least once per 18 months, during SHUTDOWN, by:

DRAFT

APR 1 1985

1. Subjecting the diesel engines to an inspection in accordance with the procedures prepared in conjunction with the manufacturer's recommendations,

2. Calibrating the diesel engine water-jacket heater temperature control functions; and the low exhaust temperature, overspeed, high water temperature, low oil pressure, and high oil temperature, "shutdown" and "declutch" protective functions; and the inlet manifold delta pressure selective function,

3. Verifying the generator capability to reject a load of greater than, or equal to, 175 KW while maintaining voltage at 480 plus or minus 48 volts and frequency at 60 plus or minus 1.2 Hz,

4. Verifying the generator capability to reject a load of 1200 KW without tripping the diesel engines. The generator voltage shall not exceed 552 volts during and following the load rejection,

5. Simulating an undervoltage relay actuation signal, and

   i. Verifying de-energization of the 480 volt A.C. essential busses and load shedding from the 480 volt A.C. essential busses, and

   ii. Verifying the diesel engines start on the auto-start signal, energize the 480 volt A.C. essential busses, energize the auto-sequenced loads, and OPERATE for greater than or equal to 5 minutes while their generator is loaded with the programmed loads. After energization, the steady state voltage and frequency shall be maintained at 480 plus or minus 48 volts and 60 plus or minus 1.2 Hz during this test.

6. Verifying that the auto-sequenced loads to each diesel generator set do not exceed 1200 KW for two diesel engines operating per generator.

7. Verifying the diesel generator's capability to be synchronized with the off-site power source upon a simulated restoration of off-site power while the generator is loaded with its emergency loads.

8. Verifying the diesel generator sets operate for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 1200 KW.

DRAFT

APR 1 1985

9. Within 5 minutes after completing a 24 hour diesel run test, verify the diesel engines start and accelerate to normal operating speed. The generator voltage and frequency shall be 480 plus or minus 48 volts and 60 plus or minus 1.2 Hz. within 10 seconds after the start signal.

d. At least once per 10 years, or after any modifications which could affect diesel generator interdependence, by starting both diesel generator sets simultaneously, during SHUTDOWN, and verifying that both diesel generator sets accelerate to normal operating speed.

e. At least once per 10 years by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or equivalent.

f. REPORTS - Diesel generator failures, as required by Table 4.8.1-2, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2. Additional reporting and requalification requirements shall be in accordance with Table 4.8.1-2.

DRAFT

APR 1 1985

TABLE 4.8.1-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests*	Test Frequency
less than or equal to 1	At least once per 31 days
greater than or equal to 2	At least once per 7 days **
greater than or equal to 3	See Table 4.8.1-2

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* Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 20 tests are determined on a per diesel generator basis. Entry into this schedule shall be made at the monthly test frequency.

** This test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 demands has been reduced to one or less. When the test is done on a once per 7 days interval, it will not be done on a STAGGERED TEST BASIS.

DRAFT

APR 1 1985

TABLE 4.8.1-2

ADDITIONAL RELIABILITY ACTIONS

No. of Failures in Last 20 Valid Tests	No. of Failures in Last 100 Valid Tests	Action
3	6	Within 30 days, prepare and maintain a report for NRC audit describing the diesel engine reliability improvement program implemented at the site. Minimum requirements for the report are indicated in Attachment 1 to this table.
5	11	If the cause of the failure cannot be found and remedied, declare the diesel generator inoperable and perform a requalification test program for the affected diesel engine. Requalification test program requirements are indicated in Attachment 2 to this table.
N/A	N/A	Submit a yearly data report on the diesel generator reliability.

DRAFT

APR 1 1985

ATTACHMENT 1 TO TABLE 4.8.1-2

REPORTING REQUIREMENT

As a minimum, the reliability improvement program report for NRC audit shall include:

a. a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

b. analysis of failures and determination of root causes of failures

c. evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

d. identification of all actions taken, or to be taken, to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

e. the schedule for implementation of each action from d) above

f. an assessment of the existing reliability of electric power to essential equipment.

Upon completion of the initial report detailing the diesel generator reliability improvement program at the site, as defined above, prepare only a supplemental report within 30 days after each failure during a valid demand for so long as the affected diesel generator unit continues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any further procedural, hardware or operational changes to be incorporated into the diesel generator improvement program and the schedule for implementation of those changes.

DRAFT

APR 1 1985

ATTACHMENT 2 TO TABLE 4.8.1-2

DIESEL GENERATOR REQUALIFICATION PROGRAM

1. Perform seven consecutive successful demands, as specified in Specification 4.8.1.1.2a), without a failure within 30 days of diesel generator being restored to OPERABLE status, and fourteen consecutive successful demands without a failure within 75 days of the diesel generator being restored to OPERABLE status.

2. If a failure occurs during the first seven tests in the requalification test program, perform seven successful demands without an additional failure within 30 days of diesel generator being restored to OPERABLE status and fourteen consecutive successful demands without a failure within 75 days of the diesel generator being restored to OPERABLE status.

3. If a failure occurs during the second seven tests (tests 8 through 14) of (1) above, perform fourteen consecutive demands without an additional failure within 75 days of the failure which occurred during the requalification testing.

4. Following the second failure during the requalification test program, be in SHUTDOWN within 36 hours.

5. During requalification testing the diesel generator should not be tested more frequently than at 24-hour intervals.

After a diesel generator has been successfully requalified, subsequent repeated requalification tests will not be required for that diesel generator under the following conditions:

a. The number of failures in the last 20 valid demands is less than 5.

b. The number of failures in the last 100 valid demands is less than 11.

DRAFT

APR 1 1985

c. In the event that following successful requalification of a diesel generator, the number of failures is still in excess of the remedial action criteria (a and/or b above) the following exception will be allowed until the diesel generator is no longer in violation of the remedial action criteria (a and/or b above):

Requalification testing will not be required provided that after each valid demand the number of failures in the last 20 and/or 100 valid demands has not increased. Once the diesel generator is no longer in violation of the remedial action criteria above, the provisions of those criteria alone will prevail.

DRAFT

APR 1 1985

AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.1 A.C. POWER SOURCES

A.C POWER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Either the Unit Auxiliary Transformer or the Reserve Auxiliary Transformer, and

b. One diesel generator set with:

1. The diesel fuel oil day tanks containing a minimum volume of 325 gallons of fuel,

2. A minimum of 10,000 gallons of diesel fuel in underground storage and an OPERABLE flow path(s) capable of transferring the fuel oil from storage to the day tank, and

3. An OPERABLE water-jacket heater for each diesel engine.

APPLICABILITY: STARTUP, SHUTDOWN and REFUELING

ACTION:

STARTUP

With any of the above minimum required electrical power sources inoperable, restore it to OPERABLE status within 6 hours or be in SHUTDOWN within the following 24 hours.

SHUTDOWN AND REFUELING

With any of the above minimum required electrical power sources inoperable, restore it to OPERABLE status within 6 hours or immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement of IRRADIATED FUEL; initiate corrective actions to restore the required sources to OPERABLE status as soon as possible.

DRAFT

SURVEILLANCE REQUIREMENTS APR 1 1985

4.8.1.2 No additional surveillance requirements are required other than those surveillances identified per Specifications 4.8.1.1.1 and 4.8.1.1.2.

DRAFT

APR 1 1985

BASES FOR SPECIFICATION LCO 3.8.1.2 / SR 4.8.1.2

The OPERABILITY of the A.C. electrical power sources and associated distribution systems during LOW POWER and POWER OPERATION ensures that sufficient power will be available, as required to perform the intended safety functions under postulated abnormal and accident conditions. The minimum specified requirements for independent and redundant A.C. electrical power sources and distribution systems are adequate to satisfy the basis of General Plant Design Criterion No. 24 as stated in Appendix C of the FSAR.

The normal A.C. electrical power source to plant auxiliaries is the Unit Auxiliary Transformer (UAT) energized by the main turbine-generator. The UAT can also be energized by the off site transmission network after the generator links have been removed to isolate the turbine generator. The UAT is connected to the 4160 volt A.C. volt busses 1 and 3.

The alternate off-site A.C. electrical power source is the Reserve Auxiliary Transformer (RAT), normally energized by the off-site transmission network. The RAT is connected to the 4160 volt A.C. Bus 2. This bus can supply, or be supplied from, the other 4160 volt A.C. busses through tie breaker connections. Upon loss-of-power from the UAT, power supply to the plant auxiliaries is transferred to the RAT.

Operation of one diesel-generator set provides for adequate electric power for one week assuming approximately 16,000 gallons of fuel oil is available. Such a reserve capacity provides ample time for obtaining additional fuel from off site sources for continued operation of the diesel-generator set. The available storage can be distributed between the diesel-generator fuel tank and two auxiliary boiler fuel oil tanks. Fuel oil transfer and/or auxiliary boiler fuel oil pumps are provided to supply the day tanks from the fuel storage tanks.

The ACTION requirements for various allowable levels of degradation of the electrical power sources provide restrictions upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources is consistent with the initial conditions/assumptions of the FSAR, and are based upon maintaining at least one of the redundant sets of on-site A.C. and D.C. electrical power sources and associated distribution systems operable during accident conditions which postulate the loss of all off-site power, compounded by a single failure of the other redundant on-site sources.

The term verify as used in the ACTION statements means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons.

DRAFT

APR 1 1985

The Surveillance Requirements are adequate to demonstrate the OPERABILITY of the off-site and on-site A.C. electrical power sources to perform their intended safety functions under postulated abnormal and accident conditions.

In particular, the surveillance requirements for the standby diesel generator sets are consistent with the intent of Regulatory Guide 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977 and Generic Letter 84-15 "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability".

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APR 1 1985

AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.2 D.C. POWER SOURCES

D.C. POWER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, three independent 125 volt D.C. electrical power sources shall be OPERABLE with:*

a. Battery Bank No. 1A and one dedicated battery charger,

b. Battery Bank No. 1B and one dedicated battery charger,

c. Battery Bank No. 1C and one dedicated battery charger,

APPLICABILITY: POWER OPERATION and LOW POWER

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 24 hours or be in at least STARTUP within the subsequent 12 hours.

b. With the above required battery chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by Specification 4.8.2.1.a)1) within 2 hours, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8.2-1 is not met, declare the battery inoperable.

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* One Battery Bank may be disconnected and the other Battery Banks cross-connected for up to 30 hours, as necessary, for the purpose of overcharging the Battery Bank.

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SURVEILLANCE REQUIREMENTS APR 1 1985

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8.2-1 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

b. At least once per 92 days, and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

1. The parameters in Table 4.8.2-1 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, and

3. The average electrolyte temperature of a representative sample of at least 20% of the cells are above 60 degrees F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, battery racks, and cell-to-cell and terminal connections show no visual indication of physical damage or abnormal deterioration, and

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material, and

3. The temperature difference of each cell-to-cell and terminal connection is less than 10 degrees F.

DRAFT

SURVEILLANCE REQUIREMENTS APR 1 1985

d. At least once per 18 months, during SHUTDOWN, by verifying that the battery capacity is adequate to supply and maintain in an OPERABLE status all of the emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during SHUTDOWN, by:

1. A performance discharge test on batteries 1A and 1B (at an average discharge rate of 85 amps) over a period of 24 hours, or until the battery terminal voltage reaches 101.5 volts. The test shall be acceptable if, after 19.2 hours the battery is capable of producing at least 85 amps, and the battery terminal voltage is greater than 101.5 volts.

2. A performance discharge test on battery 1C (at an average discharge rate of 79 amps) over a period of 12 hours, or until the battery terminal voltage reaches 101.5 volts. The test shall be acceptable if, after 9.6 hours the battery is capable of producing at least 79 amps and the battery terminal voltage is greater than 101.5 volts.

Once per 60 month interval, the performance discharge tests may be done in lieu of the battery service tests in Specification 4.8.2.1.d).

f. At least once per 18 months, during SHUTDOWN, performance discharge tests of battery capacity shall be performed on any battery that shows signs of abnormal degradation or has reached 85% of the service life expected for the application. Abnormal degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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APR 1 1985

TABLE 4.8.2-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A (1)	CATEGORY B (2)	
Parameter	Limits for each designated pilot cell.	Limits for each connected cell.	Allowable value for each connected cell. (3)
Electrolyte Level	Greater than minimum level indication mark, and less than or equal to one-quarter inch above indication mark.	Greater than minimum level indication mark, and less than or equal to one-quarter inch above indication above mark.	Above top of overflowing. plates, and not overflowing.
Float Voltage	Greater than or equal to 2.13 volts.	Greater than or equal to 2.13 volts. (c)	Greater than or equal to 2.07 volts.
Specific Gravity(a)(4)	Greater than or equal to 1.200 (b)	Greater than or equal to 1.195 volts. Average of all connected cells greater than 1.205.	Not more than .020 below the average of all connected cells. Average of all connected cells greater than 1.195 (b).

Refer to footnotes on following page.

DRAFT

APR 1 1985

1. For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

2. For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within the next 7 days.

3. Any Category B parameter not within its allowable value, declare the battery inoperable.

   a) Corrected for electrolyte temperature and level.

   b) Or battery charging current is less than 2 amps when on charge.

   c) Corrected for average electrolyte temperature.

AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.2 D.C. POWER SOURCES

D.C. POWER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following three independent 125 volt D.C. electrical power sources shall be OPERABLE:*

a. Battery Bank No. 1A and one dedicated battery charger,

b. Battery Bank No. 1B and one dedicated battery charger,

c. Battery Bank No. 1C and one dedicated battery charger,

APPLICABILITY: STARTUP, SHUTDOWN, and REFUELING

ACTION:

STARTUP

With less than the above required D.C. power sources OPERABLE, restore it to OPERABLE status within 24 hours or be in SHUTDOWN in the next 12 hours.

SHUTDOWN, REFUELING

a. With the above required battery banks inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of IRRADIATED FUEL; initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible.

b. With the above required battery chargers inoperable, demonstrate the OPERABILITY of the associated battery banks by performing Specification 4.8.2.1a)1) within 2 hours, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8.2-1 is not met, declare the battery inoperable.

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* One Battery Bank may be disconnected and the other Battery Banks cross-connected for up to 30 hours, as necessary, for the purpose of overcharging the Battery Bank.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125 volt electrical power sources shall be demonstrated OPERABLE per Specification 4.8.2.1.

DRAFT

APR 1 1985

BASES FOR SPECIFICATION 3.8.2.2 / SR 4.8.2.2

The OPERABILITY of the D.C. power sources ensures separate and normally independent sources of power for essential D.C.-powered auxiliaries and services. Each battery bank is adequate to supply shutdown DC loads for not less than one hour following a loss of all AC power. The batteries also provide the source of power, through separate inverters and static transfer switches, to the four AC instrument power buses. The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation.

The specified battery surveillances have been demonstrated through experience to provide a reliable indication of a battery cell initial breakdown well before it becomes unserviceable. Since batteries will deteriorate with time, these periodic tests will avoid precipitous failure.

The manufacturer's recommendation to equalize charge is vital to maintenance of the ampere-hour capacity of the battery. As a check upon the effectiveness of this charge, each battery is tested to verify that it meets its service ampere-hour capacity. In addition, its voltage is monitored. If a cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

DRAFT
APR 1 1985

AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION

ONSITE POWER DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open between redundant busses:

a. 480 volt A.C. Essential Bus 1 energized from the 4160 volt A.C. Bus 1.

b. 480 volt A.C. Essential Bus 2 energized from the 4160 volt A.C. Bus 2.

c. 480 volt A.C. Essential Bus 3 energized from the 4160 volt A.C. Bus 3.

d. 120 volt A.C. Non-Interruptible Busses 1A and 1A-1 energized from their associated inverter connected to D.C. Bus 1A.

e. 120 volt A.C. Non-Interruptible Busses 1B and 1B-1 energized from their associated inverter connected to D.C. Bus 1B.

f. 120 volt A.C. Non-Interruptible Busses 1C and 1C-1 energized from their associated inverter connected to Battery Bank 1C.

g. 125 volt D.C. Bus 1A energized from Battery Bank No. 1A*

h. 125 volt D.C Bus 1B energized from Battery Bank No. 1B*.

APPLICABILITY: POWER OPERATION and LOW POWER

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* One D.C. bus may be energized from an alternate source for up to 30 hours, as necessary, for the purpose of overcharging the associated battery.

DRAFT

APR 1 1985

AUXILIARY ELECTRIC POWER SYSTEMS

ACTION:

a. With one of the required 480 volt A.C. busses not energized in the required manner, reenergize the bus within 12 hours or be in at least STARTUP within the subsequent 12 hours.

b. With one A.C. Non-Interruptible bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. Non-Interruptible bus from the alternate source within 2 hours or be in at least STARTUP within the subsequent 12 hours and (2) reenergize the A.C. Non-Interruptible bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least STARTUP within the subsequent 12 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 24 hours or be in at least STARTUP within the subsequent 12 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

DRAFT

APR 1 1985

AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRUBUTION

ONSITE POWER DISTRUBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. Two 480 volt A.C. Essential Busses.

b. Two 120 volt A.C. Non-Interruptible Busses energized from their associated inverters connected to their respective Battery banks.

c. One 125 volt D.C. Buss.

APPLICABILITY: STARTUP, SHUTDOWN and REFUELING

ACTION:

STARTUP

With any of the above minimum required electrical busses not energized in the required manner, reenergize the bus as required within 12 hours or be in SHUTDOWN within the following 12 hours.

SHUTDOWN AND REFUELING

With any of the above minimum required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of IRRADIATED FUEL; initiate corrective action to reenergize the required bus as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

DRAFT

APR 1 1985

BASES FOR SPECIFICATION LCO 3.8.3. / SR 4.8.3.

The OPERABILITY and Surveillance Requirements of the onsite distribution systems ensures adequate power will be available to supply essential equipment required for the safe shutdown of the facility and to mitigate and control postulated accident conditions.

DRAFT

APR 1 1985

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.1 FUEL HANDLING AND MAINTENANCE IN THE REACTOR

LIMITING CONDITION FOR OPERATION

3.9.1 The following reactor conditions shall be maintained:

a. The PCRV shall be depressurized to atmospheric pressure or slightly below,

b. The CORE AVERAGE INLET TEMPERATURE shall be 165 degrees F or less *.

c. The reactivity of the core shall be continuously monitored by at least two neutron flux monitors, and

d. The reactor MODE shall be maintained according to the nature of the activity as follows:

1. For fuel handling-REFUELING MODE.

2. For reactor vessel internal maintenance-SHUTDOWN or REFUELING MODE.

APPLICABILITY: SHUTDOWN or REFUELING, when both primary and secondary PCRV closures are removed.

ACTION: With any of the required reactor conditions not being maintained, terminate any further fuel handling or vessel internal mantenance, retract the Fuel Handling Mechanism or any other remote operated mechanisms from the PCRV, and close the Reactor Isolation Valve or opening through the PCRV as soon as practicable.

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* Applicable only when the Fuel Handling Machine is in use and located on the reactor vessel.

DRAFT

APR 1 1985

SURVEILLANCE REQUIREMENTS

4.9.1 The reactor conditions shall be determined to be within the above limits at least once per 12 hours during refueling or internal reactor vessel maintenance:

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.9.1 / SR 4.9.1

To prevent the outleakage of primary coolant and potential release of activity during refueling or maintenance in the reactor vessel, the reactor must be depressurized and maintained within the required conditions. The CORE AVERAGE INLET TEMPERATURE is limited to 165 degrees F to prevent pressurization of the fuel handling equipment over 5 psig (the maximum allowable working pressure of the fuel handling equipment) as a result of accidental inleakage of water into the vessel during refueling.

DRAFT

APR 1 1985

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.2 FUEL HANDLING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.2 The Fuel Handling Machine shall be OPERABLE with the following conditions met:

a. The pressure in the Fuel Handling Machine at atmospheric pressure or slightly below.

b. The Fuel Handling Purge System shall be connected and OPERABLE, with a supply of helium available, and the Gas Waste System available to receive purge gas.

c. One cooling water coil shall be OPERATING with the outlet coolant water temperature 150 degrees F or less, and

1. An additional cooling water coil OPERABLE, or

2. The backup fire water connections and hose OPERABLE.

d. The reactor building overhead crane attached, unless the Fuel Handling Machine is bolted to a Reactor Isolation Valve over the reactor, over a Fuel Storage Well, or over the Fuel Shipping Cask Loading Port.

APPLICABILITY: During REFUELING (including use of the Fuel Handling Machine for reactor internal maintenance and any handling of irradiated fuel).

ACTION:

1. With requirements a, b, or d above not satisfied, within one hour initiate action to retract the Fuel Handling Machine to its uppermost position and close the Reactor Isolation Valve and Fuel Handling Machine Cask Valve.

2. With requirement a above not satisfied when the Fuel Handling Machine Cask Valve is closed, within one hour initiate action to reduce gas inflow to the Fuel Handling Machine and/or increase purge gas outflow to the gas waste system, as appropriate.

DRAFT

APR 1 1985

3. With requirement c above not satisfied, within one hour initiate action to return the irradiated fuel elements within the Fuel Handling Machine to the reactor core or to the Fuel Storage Facility, and then terminate fuel handling with the Fuel Handling Machine.

SURVEILLANCE REQUIREMENTS

4.9.2 Within 31 days prior to REFUELING the Fuel Handling Machine shall be demonstrated OPERABLE by:

a. Functionally testing the Fuel Handling Machine, Fuel Handling Machine Cask Valve and Reactor Isolation Valves, interlocks, limit switches, and alarms.

b. Functionally testing the redundant cooling water coils.

c. Calibrating and functionally testing the cooling water flow and temperature alarms.

d. Functionally testing the Fuel Handling Machine Cooling Water Leak Detector.

e. Visually inspecting the backup fire water connections and hose.

f. Functionally testing the Fuel Handling Purge System, including availability of a supply of helium and the availability of the Gas Waste System to receive purge gas.

g. Establishing the OPERABILITY of the Reactor Building Crane and the Fuel Handling Machine special lifting device prior to attachment to fuel handling equipment.

BASIS FOR SPECIFICATION LCO 3.9.2 / SR 4.9.2

The OPERABILITY requirements for the Fuel Handling Machine ensure that:

1. Failure of a primary seal at any boundary would result in negligible outleakage or no more than slow inleakage of air, since the machine is maintained at atmospheric pressure or slightly below.

2. The Fuel Handling Machine is permanently connected to the gas waste system, and can be purged with helium while it is in transit and/or when the cask valve is shut, giving assurance that no pressure buildup in or escape of fission products from the machine should occur.

3. Fuel elements contained in the Fuel Handling Machine will be maintained at surface temperatures below 750 degrees F to prevent significant graphite oxidation in the event there is any inleakage of air. An outlet cooling water temperature of less than or equal to 150 degrees F from one of the two redundant cooling systems provides adequate cooling to maintain the fuel elements below 750 degrees F (see FSAR Section 14.6.3.1).

4. A seismic event will not cause the Fuel Handling Machine to topple.

The objective of this specification is to ensure the prevention of any uncontrolled release of radioactivity during fuel handling.

The Fuel Handling Machine provides for the safe refueling of the reactor. To assure the reliability of the Fuel Handling Machine during the refueling operation, the machine and the isolation valve(s) will be functionally tested prior to refueling.

A functional test of the Fuel Handling Purge System and cooling water system and an inspection of the backup Firewater System connection and hose will be made. These checks are to assure the capability to maintain the proper atmosphere environment within the machine to prevent any uncontrollable release of activity, and the capability to maintain temperature of fuel elements within the machine below 750 degrees F.

DRAFT

APR 1 1985

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.3 FUEL STORAGE WELL

LIMITING CONDITION FOR OPERATION

3.9.3 Each Fuel Storage Well containing irradiated fuel shall be maintained at atmospheric pressure or below, and

a. Both cooling water coils shall be OPERABLE with at least one coil IN OPERATION, and the outlet cooling water temperature of each operating coil shall be 150 degrees F or less, or

b. One cooling water coil shall be OPERATING with the outlet cooling water temperature 150 degrees F or less, and the Fuel Storage Well Emergency Booster Fan OPERABLE and capable of moving a minimum total air flow of 9,000 cfm through the Fuel Storage Facility.

APPLICABILITY: During storage of irradiated fuel in the Fuel Storage Well.

ACTION: With less than the above required conditions satisfied, within 1 hour initiate corrective action to re-establish the required conditions within 24 hours; or

a. Transfer IRRADIATED FUEL to a storage well or wells for which the required conditions are met, or

b. Establish backup cooling to the affected fuel storage wells. Prior to reaching 750 degrees F for the hottest fuel element.

SURVEILLANCE REQUIREMENTS

4.9.3 The cooling water, purge, and ventilation systems for each Fuel Storage Well containing IRRADIATED FUEL shall be demonstrated OPERABLE:

a. At least once per 24 hours by:

1. Verifying through station log review or other information that:

a) Neither cooling water coil inoperable with at least one coil IN OPERATION, and the outlet cooling water temperature is 150 degrees F or less, or

b) When only one cooling water coil is OPERABLE,

1) The outlet cooling water temperature of the operating coil is 150 degrees F or less, and

2) The Fuel Storage Well Emergency Booster Fan is inoperable not out of service.

2. Verifying that the pressure within the well is at atmospheric pressure or slightly below.

b. At least once per 31 days by:

1. Verifying that both cooling water coils are OPERABLE.

2. Verifying that the Fuel Storage Well Emergency Booster Fan operates upon manual initiation.

c. At least once per 12 months by:

1. Calibrating and functionally testing the Fuel Storage Facility helium pressure indicators and alarms.

2. Calibrating and functionally testing the Fuel Storage Facility cooling water system flow indicators, and flow and temperature alarms.

3. Functionally testing the capability of the Fuel Storage Well Emergency Booster Fan to draw a minimum of 9000 cfm of air through the Fuel Storage Facility.

BASIS FOR SPECIFICATION LCO 3.9.3/ SR 4.9.3

The storage well cooling water system is designed with two 100% capability cooling coils supplied from independent water sources (see FSAR Section 9.1.2).

The accident conditions described in the FSAR postulate the total loss of water cooling to one of the nine Fuel Storage Wells. If this were to occur, adequate cooling could be achieved by an increase in the normal ventilation air flow to cool the well by convection on the external surface. The increase in air flow is supplied by the Fuel Storage Well Emergency Booster Fan. This Specification is based on the analysis in FSAR Section 14.6.3.2 which uses the conservative assumption that a total flow of only 9000 cfm would be drawn (equally divided) through all three vault compartments of the Fuel Storage Facility, thus adequately protecting the affected storage well and fuel within it from damaging temperatures. (There are three Fuel Storage Wells in each of the three Fuel Storage Vaults). The specified test and testing frequency are sufficient to demonstrate the operability of the Fuel Storage Well Emergency Booster Fan, should it be called upon for performance of its required safety function.

To prevent significant oxidation of the irradiated fuel, the Fuel Storage Wells are designed to maintain the irradiated fuel cool and under a nominally dry atmosphere of helium. All conditions connected with this requirement are monitored by pressure, temperature, and flow sensitive devices. The temperature and flow detecting devices maintain surveillance of the wells' two independent cooling water systems and are set to alarm at previously determined maximum or minimum values. The 150 degrees F or less outlet temperature of water from a single cooling coil assures the hottest irradiated fuel elements will be maintained below 750 degrees F, preventing any significant graphite oxidation in the event of air inleakage into the storage well. The Helium Storage System provides purified helium for this service, giving sufficient protection against a moist atmosphere. Over-pressurization of a storage well is alarmed to the operator and additional protection is provided by relief valves. The specified annual surveillance interval is sufficient to ensure proper operation of the instrumentation and to verify adequate conditions.

DRAFT

APR 1 1985

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.4 SPENT FUEL SHIPPING CASK

LIMITING CONDITION FOR OPERATION

3.9.4 A Spent Fuel Shipping Cask shall not be loaded or partially loaded with irradiated fuel having less than 100 days of fission product decay, nor shall it be transported off-site if the radiation level at any point on the accessible external surface of the cask exceeds 200 millirem per hour.

APPLICABILITY: At all times

ACTION:

a. Irradiated fuel having less than 100 days of fission product decay shall be removed from the Spent Fuel Shipping Cask prior to lifting of the cask.

b. Irradiated fuel which results in a radiation level greater than 200 millirem per hour at any point on the accessible external surface of the Spent Fuel Shipping Cask, shall be removed from the cask prior to shipment off-site.

c. The provisions of Specifications 3.0.5 and 3.0.6 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4

a. Prior to loading the Spent Fuel Shipping Cask, the irradiated fuel elements shall be determined to have undergone fission product decay for 100 days or more.

b. A radiation survey of the accessible external surface of Spent Fuel Shipping Cask shall be made prior to transport of the cask off-site, to determine compliance with the above radiation limit.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.9.4 / SR 4.9.4

The potential radiological consequences of an accident whereby the Spent Fuel Shipping Cask breaks open while being lowered to the truck, have been analyzed (FSAR Section 14.6.3.3) assuming that the cask is loaded with the most radioactive spent fuel elements contemplated to be shipped from the plant after 100 days of fission product decay. Determination prior to loading in the cask that the fuel elements have undergone at least 100 days of fission product decay ensures that cask lifting within the plant is bounded by the safety analysis.

The radiation dose limit of 200 mrem/hr at the outer surface of the cask is in accordance with 10CFR 71 external radiation standards for all packages. Performace of a radiation survey of the external suface of the cask prior to shipment assures that the cask, when shipped, will be in compliance with this requirement.

DRAFT

APR 1 1985

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.5 COMMUNICATIONS DURING CORE ALTERATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct two-way communications shall be maintained between the Operations Control Room personnel and personnel at the Fuel Handling Machine (FHM) Control Room.

APPLICABILITY: During CORE ALTERATIONS

ACTION:

When direct communications between the Operations Control Room personnel and personnel at the FHM Control Room cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the Operations Control Room and personnel at the FHM Control Room shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

DRAFT

APR 1 1985

BASIS FOR SPECIFICATION LCO 3.9.5 / 4.9.5

The requirement for communications capability ensures that refueling personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS, and Operations Control Room personnel can be informed by refueling personnel whenever CORE ALTERATIONS are being performed such that close monitoring of core conditions is performed during alterations.

The FHM Control Room and Operations Control Room personnel must coordinate control rod movements to ensure the required shutdown margin is maintained during CORE ALTERATIONS. Continuous communication also permits the Operations Control Room to immediately request a stop of movements causing excessive count rate changes.

The surveillance times specified give adequate assurance that the communications will be available as needed, since there will be continuous communication maintained during CORE ALTERATIONS.

DRAFT
APR 1 1985

SPECIAL TEST EXCEPTIONS

3/4.10.1 XENON STABILITY

LIMITING CONDITION FOR OPERATION

3.10.1 The requirements of Specification 3.1.4 may be suspended during the performance of the Xenon Stability physics tests provided:

a. Power perturbations beyond that which are expected do not occur, and

b. The limits per Specification 3.2.2 are maintained, and

c. A detailed engineering evaluation is performed in advance of the test and reviewed by the NFSC to assure that the intent of Specifications 3.1.4 and 3.2.2 are met throughout the period that Specification 3.1.4 requirements are suspended.

APPLICABILITY: POWER OPERATION*

ACTION: With the occurrence of unexplained power perturbations or if the limits of Specification 3.2.2 are exceeded while Specification 3.1.4 is being suspended, either:

a. Reduce THERMAL POWER sufficiently to stop power perturbations or satisfy the requirements of Specification 3.2.2, or

b. If only the limits of Specification 3.2.3 are exceeded, REGION PRIMARY COOLANT FLOW may be increased sufficiently to satisfy the requirements of LCO 3.2.3.

SURVEILLANCE REQUIREMENTS

4.10.1 The above required Xenon Stability Testing shall be performed per the sequence 75 B-0 Start-up Test.

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* During Xenon Stability Physics Tests

BASIS FOR SPECIFICATION LCO 3.10.1 / SR 4.10.1

In order for the Xenon Stability characteristics of the reactor core to be demonstrated, it is necessary to initiate power redistributions by inserting an out of sequence rod. The limits on the average temperature rise from circulator inlet to core outlet contained in Specification 3.2.2, shall be maintained throughout this test. Maintaining operation within these limits will fully protect plant equipment and the health and safety of the public.

The temperature rise across each refueling region shall be determined at least once per hour during the physics tests in which Specification 3.1.4 is suspended and shall be determined to be within the limits specified in Specification 3.2.2.

A detailed engineering evaluation will be performed in advance of any Xenon Stability Testing in order to analyze the planned test configuration to assure compliance with the intent of Specification 3.1.4 and Specification 3.2.2 throughout the test evolution. This engineering evaluation will assure that the maximum worth rod pair criteria of 0.012 delta k is met (see FSAR, Section 14.2.2.6 and 3.5.5.1), and that the assumptions utilized in development of core safety limit Specification 2.1 (with regards to region, column, and axial peaking factors) remain valid throughout the test.

DRAFT

APR 1 1985

SECTION 5.0

DESIGN FEATURES

DRAFT

APR 1 1985

5.1 SITE

5.1.1 The Fort St. Vrain Nuclear Generating Station, Unit No. 1, is situated on a tract of land located about 3.5 miles northwest from the center of Platteville, Colorado. The tract is situated in Weld County, Colorado (See FSAR Section 2.1).

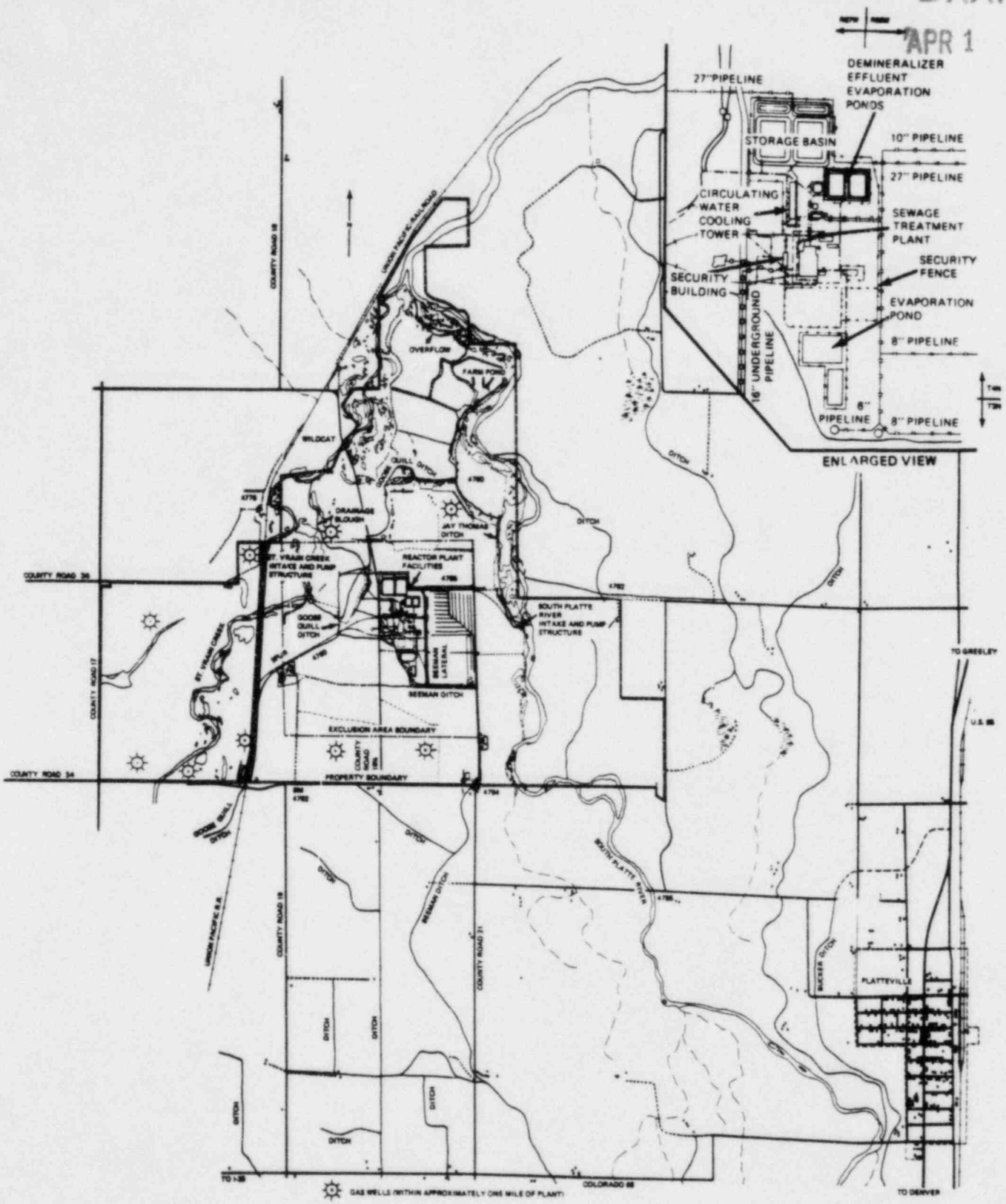
The exclusion area is approximately 1 mile square and is defined in Figure 5.1-1. The closest distance from the reactor building to the boundary of the exclusion area is 1,935 feet. The limits of 10 CFR 20 shall apply at the boundary of this exclusion area. The Low Population Zone (LPZ) is defined by a radius of 16,000 meters. The exclusion area is zoned industrial, and the area surrounding the exclusion area is zoned agricultural. Agricultural activities may continue on the site including a portion of the exclusion area, and an evacuation procedure will be maintained. There are no permanent residences located within the exclusion area.

An Information Center is located within the exclusion area, but outside the main gate. An evacuation procedure will be maintained for the Information Center.

Points where radioactive gaseous and liquid effluents are released are shown on Figure 5.1-2 as are the liquid effluent pathways leaving the site.

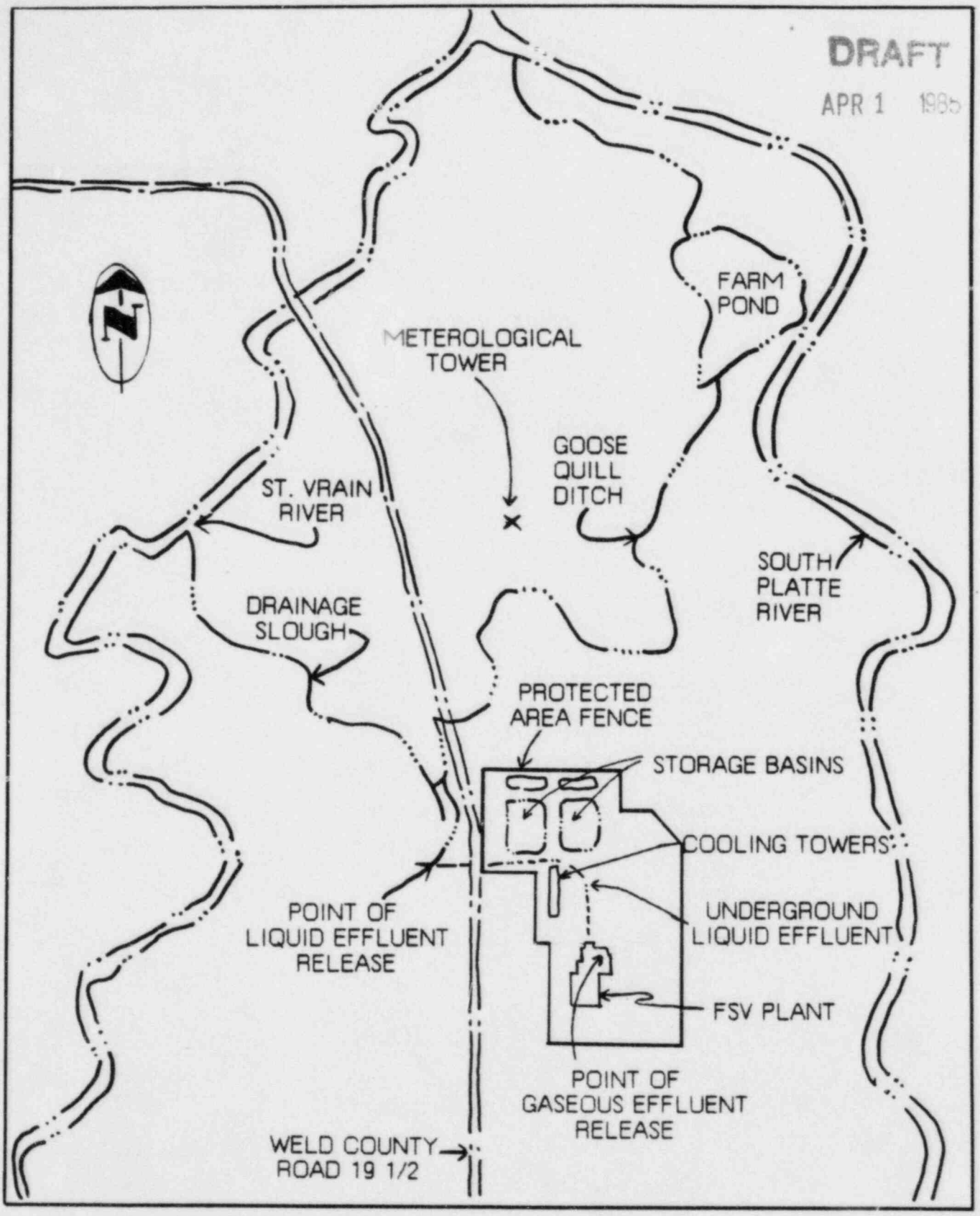
DRAFT

APR 1 1980



Site of the Fort St. Vrain
Nuclear Generating Station

Figure 5.1—1



Fort St. Vrain Site Detail

Figure 5.1–2

DRAFT

APR 1

5.2 REACTOR COOLANT SYSTEM AND STEAM PLANT SYSTEM

PRESTRESSED CONCRETE REACTOR VESSEL (PCRV)

5.2.1 The PCRV is constructed of high strength concrete reinforced with bonded reinforcement steel and prestressed with steel tendons. Prestressing tendons, located in conduits embedded in the concrete are used to prestress the entire structure. Access to the tendons is provided so that most tendons can, if necessary, be retensioned or selectively removed for inspection and replaced. The type and number of tendons in the PCRV are shown below:

Type of Tendon	Number of 1/4" Wires per Tendon	Number of Tendons
Longitudinal	169	90
Circumferential		
a) Head	169	100
b) Wall	152	210
Bottom Cross Head	169	24
Top Cross Head	169	24

The temperature of the PCRV concrete is controlled by means of insulation mounted on the inside surface of the liner, and cooling tubes welded to the concrete side of the liner. The whole of the internal surface of the liner is covered by the thermal barrier which uses Kaowool insulation, a ceramic fiber blanket material of high chemical purity which is about 50% alumina and 50% silica.

DRAFT

APR 1 1985

The various PCRV penetrations required for refueling, maintenance, control rods or operation of circulators and steam generators are provided with liners that are welded to the cavity liner and extend through the concrete to provide leak-tight access to the reactor internals. Each penetration is provided with two closures in series, a primary closure and a secondary closure. The primary closures, together with the PCRV structure, contain the radioactive primary coolant in a manner analogous to a conventional primary vessel. The secondary closure encloses the primary closure and contains the radioactive primary coolant that might be released from a leaking primary closure in a manner analogous to a conventional secondary containment. The primary and secondary closures are similar to conventional pressure vessel closures and are flat or formed heads. The closures incorporate elastomer or metallic seals, depending on the operating temperatures, and are attached to the penetration liner flanges with bolts or shear rings. Independently anchored flow restriction devices are in the larger penetrations to limit the flow rate of primary coolant if both primary and secondary closures were to fail completely and instantaneously.

STEAM GENERATOR ORIFICES

5.2.2 The steam generator modules are provided with two sets of orifices:

a) The variable feedwater ringheader trim valves, which include mechanical stops to prevent total closure.

b) The fixed feedwater orifices in the economizer tube inlet subheaders.

These flow limiting devices are provided to limit water/steam inleakage to the primary coolant through a subheader tube rupture as described in FSAR Section 6.4 and Section 14.5.

DRAFT

APR 1 1985

DESIGN FEATURES

STEAM SAFETY VALVES

5 2.3 The steam plant contains the following steam pressure safety valves, with set pressures and capacities as shown below:

Valve	Quantity	Set Press. psig	Type and Capacity
Bypass Flash Tank	6	975 to 1020	ASME Code, Section VIII, Spring-Loaded Valves; 105% Plant Capacity (Total)
Hot Reheat	6	700 to 735	ASME Code, Section VIII, Spring-Loaded Valves; 105% Plant Capacity (Total)

The bypass flash tank and hot reheat line safety valves prevent over-pressure of the cold reheat and the hot reheat piping, respectively. As long as either the cold reheat or the hot reheat block valves are open, these valves also prevent over-pressure of the reheaters. These valves discharge to atmosphere.

DRAFT
APR 1 1985

5.3 REACTOR CORE

5.3.1 Reactor Assembly

The reactor core consists of: (1) removable fuel elements which contain the fuel (U & Th), the moderator (graphite) and burnable poison (boron), and (2) radial and axial reflectors which consist of removable reflector elements and permanent blocks which are made of graphite, and in some cases incorporating boron or structural steel. The reactor core assembly, including reflector, has an overall assembly height of about 23.9 feet and a diameter of about 27.3 feet. The approximate weight of the core assembly is 1,348,000 pounds. The preceding description includes the core support graphite blocks.

The reactor reactivity control consists of 37 pairs of control rods containing boron carbide, which are supplemented by burnable poison (boron) in selected fuel elements as required. A reserve shutdown system consisting of 37 hoppers of boron carbide-graphite balls is also provided.

A variable orifice flow-control assembly is located at the inlet to each of the 37 refueling regions to provide adjustment of the coolant flow through the region.

5.3.2 Active Core

The active core consists of 1482 hexagonal graphite fuel elements stacked in 247 vertical fuel columns. The fuel elements form the active core which is essentially a right circular cylinder 15.6 feet in height and 19.5 feet in equivalent diameter. The active core is completely surrounded by a graphite reflector. Within the core array, the fuel columns are grouped into 37 refueling regions containing seven fuel columns each, except for six outer corner regions which contain five fuel columns each.

The center fuel column of each of the 37 fuel regions is a control rod column. Each control rod column contains two control rod channels and one reserve shutdown absorber material channel. Each control rod channel has a diameter of 4.0 inches and the two channels have a centerline pitch spacing of 9.7 inches. The reserve absorber material shutdown channel has a diameter of 3-3/4 inches. The control rod channels are continuous from the top face of the top reflector and terminate in the bottom reflector at an elevation not greater than 27.0 inches above the top face of the core support block. The reserve shutdown absorber channel is continuous from the top face of the top reflector and terminates in the bottom fuel element at an elevation not greater than 47.5 inches above the top face of the core support block.

Each fuel element is a hexagonal right prism with nominal dimensions of 14.2 inches across the flats by 31.2 inches high. The fuel beds and coolant channels are distributed on a triangular array of about 3/4 inch pitch spacing with an ideal ratio of two fuel beds for each coolant channel. The bottom of the fuel beds in the bottom fuel element of the control rod fuel column does not exceed a length of 23.1 inches from the top face of the fuel element.

5.3.3 Fuel

The fuel consists of fissile uranium highly enriched (93.15%) in uranium 235 and fertile thorium. The initial fuel loading is about 773 Kg of uranium and 16,000 Kg of thorium. The initial core is loaded with 13 fuel compositions whose distribution within the core is designed to mock up the fuel content of the equilibrium cycle refueling regions and to shape the radial and axial power distribution. Fuel is designed for up to a six operating cycles. About one-sixth of the core will be replaced at each refueling interval. The fuel loading in a reload segment will be about 200 Kg of uranium and 2300 Kg of thorium.

All uranium and thorium in the reference fuel elements is in the form of heavy metal carbide and pyrocarbon, referred to as coated fuel particles. The coatings form the primary fission product barrier. The coated fuel particles consist of two general types, fissile particles (TH:UC/2) and fertile (TH C/2) particles. The fissile particles shall contain thorium and uranium in a weight ratio of about 3.6 to 1 (+1.2, - 0.2) of thorium to uranium. The fertile particles shall contain only thorium.

In addition to the reference fuel elements, eight test fuel elements are included in the reactor core. These eight test elements (FTE1-8) contain small quantities of test fuel particles that are in various ways different from the reference fuel. The description of the test fuel elements is contained in Table 5.3-1.

The coated fuel particles are bonded together with a carbonaceous material to form fuel rods. The fuel rods are completely surrounded and contained by graphite which forms the structural part of the fuel element and, in addition to the carbon contained within the fuel rods, also serves as the sole moderator. The reference fuel elements are fabricated from H-327 needle coke (anisotropic) graphite, as described in the Fort St. Vrain FSAR, Section 3.0. The test fuel elements are fabricated from H-451 near-isotropic graphite in anticipation of qualifying this material for future use in all reload fuel for the reactor.

DRAFT

APR 1 1985

Beginning with core Segment 9 (Reload 3), H-451 near-isotopic graphite is used in the fabrication of reload fuel elements in addition to or in place of the previous reference H-327 needle coke (anisotropic) graphite.

5.3.4 Reload Segment Design

Each reload segment comprises about one-sixth of the reactor core. Consequently, the reactor core after a refueling consists of six segments with different degrees of core burnup scattered throughout the core. In addition, the burnable poison being added for reactivity control is only present within the new fuel elements. As a consequence, each of the 37 core regions has a different effective multiplication constant, k (eff).

To satisfy the criteria for reactor power distribution and maximum control rod worth, each refueling cycle has an analytically justified control rod withdrawal sequence.

The following criteria shall be used as the basis to establish any control rod withdrawal sequence:

a. The maximum calculated reactivity worth of any rod pair in any normal operating rod configuration with the reactor critical shall not exceed 0.047 delta k.

b. The maximum allowable calculated single control rod pair worth, at any core condition, during power operation shall depend on the available core temperature coefficient. The accidental removal of the maximum worth single rod-pair shall result in a transient with consequence no more severe than the withdrawal of .012 delta k, at rated power, from a core which has a reactivity defect between 220 degrees F and 1500 degrees F of .028 delta k. (FSAR, Section 3.5.5.1 and 14.2.2.6).

c. Calculated power peaking factors in any normal operating rod configuration shall be within the following specified range:

Region Peaking Factor = Average Region P/Average Core P

CORE AVERAGE OUTLET TEMP.	Core Region Peaking Factor
Greater than or equal to 1250 degree F	Between 0.4 and 1.83
Between 950 and 1250 degrees F	Between 0.4 and 2.15
Less than 950 degrees F	Between 0.28 and 3.00

Axial Peaking Factor = Average Lower Layer P/Average Region P

CORE AVERAGE OUTLET TEMP.	Applicable Regions	Axial Peaking Factor
Greater than 220 degrees F	Regions with control rods fully inserted or withdrawn	Less than or equal to 0.90
	Regions with control rods partially inserted more than two Feet into the core	Less than or equal to 1.23

DRAFT

APR 1 1985

Intra-Region Peaking = Average Column P/Average Region P

CORE AVERAGE OUTLET TEMP.	Applicable Regions	Intra-region Peaking Factor
Greater than or equal to 950 degrees F	Regions with control rods fully inserted	Less than or equal to 1.46
Greater than or equal to 950 degrees F	Regions with control rods partially inserted more than two feet into the core	Less than or equal to 1.40
Greater than or equal to 950 degrees F	Regions with control rods not inserted more than two feet into the core	Less than or equal to 1.34
Less than 950 degrees F	All regions	Less than or equal to 1.61

The reserve shutdown system has been designed to ensure that the reactor can be shutdown to refueling conditions even in the incredible event that all control rods fail to insert. After extended operation, it must cover the reactivity temperature defect for core cooldown, the decay of Xe-135, and some decay of PA-233 minus the reactivity effect of Sm-149 buildup.

Since the reactivity worth of the temperature change (core cooldown) and the reactivity worth of the highest worth single reserve shutdown units vary for each cycle, calculations for each reload segment must demonstrate that the reserve shutdown system is capable of achieving its design requirement.

DRAFT

APR 1 1985

Analysis shall demonstrate that the Reserve Shutdown System has sufficient reactivity worth to achieve a minimum shutdown margin of 0.01 delta k for the following conditions;

a. Core cooldown to CORE AVERAGE TEMPERATURE of 220 degrees F.

b. Full Xe decay and one week of Pa decay including Sm buildup.

c. The highest worth reserve shutdown unit in each subset has not been inserted.

d. All control rods have failed to insert.

5.3.5 Reflector

Reflector elements above, below, and immediately adjacent to the side of the active core are hexagonal right prisms with nominal dimensions of 14.2 inches across flats and 15.6, 23.4, or 31.2 inches high, as required. The outer peripheral envelope of the reactor core reflector graphite contains boron to minimize the neutron flux leaving the reflector. The side reflector contains nominal 2.3 weight percent boron stainless steel pins within the spacer blocks. The middle layer of lower reflector elements, excluding the central element in each core region, contains 25 weight percent boronated graphite pellets enclosed in hastalloy-X cans. The top layer of reflector above the hexagonal columns contains 1 weight percent crushed boronated graphite. The top layer of reflector above the permanent side reflector blocks contains 1 weight percent boronated graphite enclosed in steel cans.

TABLE 5.3-1

DESCRIPTION OF FORT ST. VRAIN FUEL ELEMENTS FTE-1 THROUGH FTE-8

	FTE-1	FTE-2	FTE-3	FTE-4	FTE-5	FTE-6	FTE-7	FTE-8
Segment	2	3	4	5	6	7	7	7
Graphite Type	H-451	H-451	H-451	H-451	H-451	H-451	H-451	H-451
Fissile Fuel Type	UC_2 TRISO	UC_2 TRISO plus test fuel (a)	UC_2 TRISO	UC_2 TRISO plus test fuel (a)	UC_2 TRISO	UC_2 TRISO plus test fuel (a)	$(Th,U)C_2$ TRISO	$(Th,U)C_2$ TRISO
Fertile Fuel Type	ThO_2 TRISO	ThO_2 TRISO plus BISO	ThO_2 TRISO	ThO_2 TRISO plus BISO	ThO_2 TRISO	ThO_2 TRISO plus BISO	ThC_2 TRISO	ThC_2 TRISO
Method of Fuel Rod Curing[b]	CIP	CIP	CIP	CIP	CIP	CIP	CIB	CIB
Residence Time, Equivalent Full Power Days	200	500	800	1100	1400	1700	1700	1700

(a) Test fuel includes:

$(Th,U)C_2$ TRISO,	88 rods per element CIB	HEU
UC_xO_y * TRISO/ThO_2 BISO,	350 rods per element CIP	HEU
UC_2 TRISO/ThO_2 TRISO,	176 rods per element CIP	HEU
(Th,U)O TRISO/ThO TRISO,	88 rods per element CIT	MEU

(b) CIP = cure-in-place fuel rod carbonization; CIB = cure in alumina bed - reference FSV process; CIT = cure-in-tube, graphite crucibles, simulating conditions as experienced in cure-in-place.

* X and y represent the mean quantities of carbon and oxygen and do not signify a specific compound. These values will be explicit in the final fuel specification. All kernels of this type are derived from WAR beads.

DRAFT
APR 1 1985

DRAFT
APR 1 1985

5.4 FUEL STORAGE

CRITICALITY

5.4.1 The fuel storage wells are designed and shall be maintained to preclude criticality when completely filled with fuel and flooded with water. The multiplication factor for the worst flooding situation shall be less than 0.85 as described in Section 9.1.2.3 of the FSAR.

CONTAINMENT

5.4.2 The fuel storage wells are designed and shall be maintained to have gas- and water-tight containment.

NEUTRON ABSORBER-HEAT SINK

5.4.3 The fuel storage wells are designed and shall be maintained to have a space between the inner tank and outer cylindrical shell filled with a granular material to act as a built-in neutron absorber and heat sink.

5.5 METOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-2.

5.6 COMPONENT AND TRANSIENT CYCLIC LIMIT

5.6.1 The components identified in TABLE 5.6-1 are designed and shall be maintained within the cyclic limits of TABLE 5.6-1 when evaluated against the identified plant transients of TABLE 5.6.1.

DRAFT
APR 1 1985

Table 5.6-1

ALLOWED NUMBER OF COMPONENT TRANSIENTS

COMPONENT	DC-5-2 TRANSIENT NUMBER (1)					
	9	5	2a, 2b or 8	5, 7 or 9	6	5,6,7 or 9
Steam Generator External feedwater pipe External	470	700				
main steam pipe Internal			1000			
main steam bypass piping -8" X 16"latrolet				123		
-bypass line			550		175	
Hot reheat steam piping			560			185

(1) DC-5-2 Transient identification

2a- Shutdown to refueling status

2b- Shutdown with full helium inventory

5 - Reactor scram

6 - Turbine trip

7 - Primary system depressurization

8 - Loop shutdown

9 - Steam leak with loop dump

DRAFT
APR 1 1985

SECTION 6.0

ADMINISTRATIVE CONTROLS

DRAFT

 APR 1 1985

INDEX

DRAFT

APR 1 1985

6.5.2 NUCLEAR FACILITY SAFETY COMMITTEE (NFSC)

FUNCTION
COMPOSITION
ALTERNATES
CONSULTANTS
MEETING FREQUENCY
QUORUM
REVIEW
AUDITS
AUTHORITY
RECORDS

6.6 REPORTABLE EVENTS ACTION

6.7 SAFETY LIMIT VIOLATIONS

6.8 PROCEDURES AND PROGRAMS

6.9 REPORTING REQUIREMENTS

6.9.1 ROUTINE REPORTS AND REPORTABLE EVENTS

STARTUP REPORT
ANNUAL REPORTS
SEMI-ANNUAL REPORTS
MONTHLY OPERATING REPORTS
REPORTABLE EVENTS
PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP
THIRTY DAY WRITTEN REPORTS
NON-ROUTINE RADIOLOGICAL REPORTS

6.9.2 SPECIAL REPORTS

6.10 RECORD RETENTION

6.11 RADIATION PROTECTION PROGRAM

6.12 HIGH RADIATION AREA

6.13 PROCESS CONTROL PROGRAM (PCP)

DRAFT
APR 1 1985

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15 MAJOR CHANGES TO LIQUID, GASEOUS AND SOLID RADWASTE TREATMENT SYSTEMS

6.16 FUEL SURVEILLANCE PROGRAM

DRAFT

APR 1 1985

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the Shift Supervisors office, a designated individual) shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President of Electric Production shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

MANAGEMENT

6.2.1 The management organization for administrative and departmental support shall be shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. A licensed operator must be in the Control Room at all times when fuel is in the reactor. During reactor STARTUP, SHUTDOWN, and recovery from reactor trip, two licensed operators must be in the Control Room.

c. An operator or technician qualified in radiation protection procedures shall be present at the facility at all times that there is fuel on site.

d. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to Fuel Handling who has no other concurrent responsibilities during this operation.

DRAFT

APR 1 1985

e. A site Fire Brigade of at least five members shall be maintained on site at all times*. The Fire Brigade shall not include three (3) members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

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* Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed two hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

DRAFT
APR 1 1985

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

Position	Number of Individuals Required to Fill Position	
	During SHUTDOWN, Refueling Shutdown (a)	All Other Conditions
SS (SRO)	1	1
SRO	Not Required	1
RO	1	2 (b)
EO	1	1
AT	Not Required	1
TA (c)	1	1

SS - Shift Supervisor with a Senior Reactor Operator's License
SRO - Individual with a Senior Reactor Operator's License
RO - Individual with a Reactor Operator's License
EO - Equipment Operator
AT - Auxiliary Tender
TA - Technical Advisor (on 1-hour call)

In addition to the requirements of Section 7.1.2, the following apply:

During SHUTDOWN, or REFUELING MODES conditions, an individual (other than the Technical Advisor) with a valid RO (or SRO) license shall be present in the Control Room.

During all other conditions, an individual (other than the Technical Advisor) with a valid SRO license shall be present in the Control Room.

DRAFT

APR 1 1985

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. This provision also does not exclude the requirement for an SRO licensed individual to be in the Control Room at all times other than the SHUTDOWN conditions specified in Table 6.2-1.

During any absence of the Shift Supervisor from the control room while the unit is in POWER OPERATION, LOW POWER or STARTUP, an individual (other than the Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in SHUTDOWN or REFUELING, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

NOTES:

a. Per Technical Specification definitions, Section 1.0.

b. One of the two Reactor Operators may be an Equipment Operator with a valid RO license provided that the staffing requirement for Equipment Operators is being met by another individual qualified as an Equipment Operator.

c. The Technical Advisor shall be scheduled as required by Specification 6.2.3.2(5).

DRAFT
APR 1 1985

FORT ST. VRAIN NUCLEAR GENERATING STATION
ADMINISTRATIVE AND DEPARTMENTAL MANAGEMENT ORGANIZATION CHART

NFSC - Nuclear Facility Safety Committee
PORC - Plant Operations Review Committee

** Serves as the Radiation Protection Manager

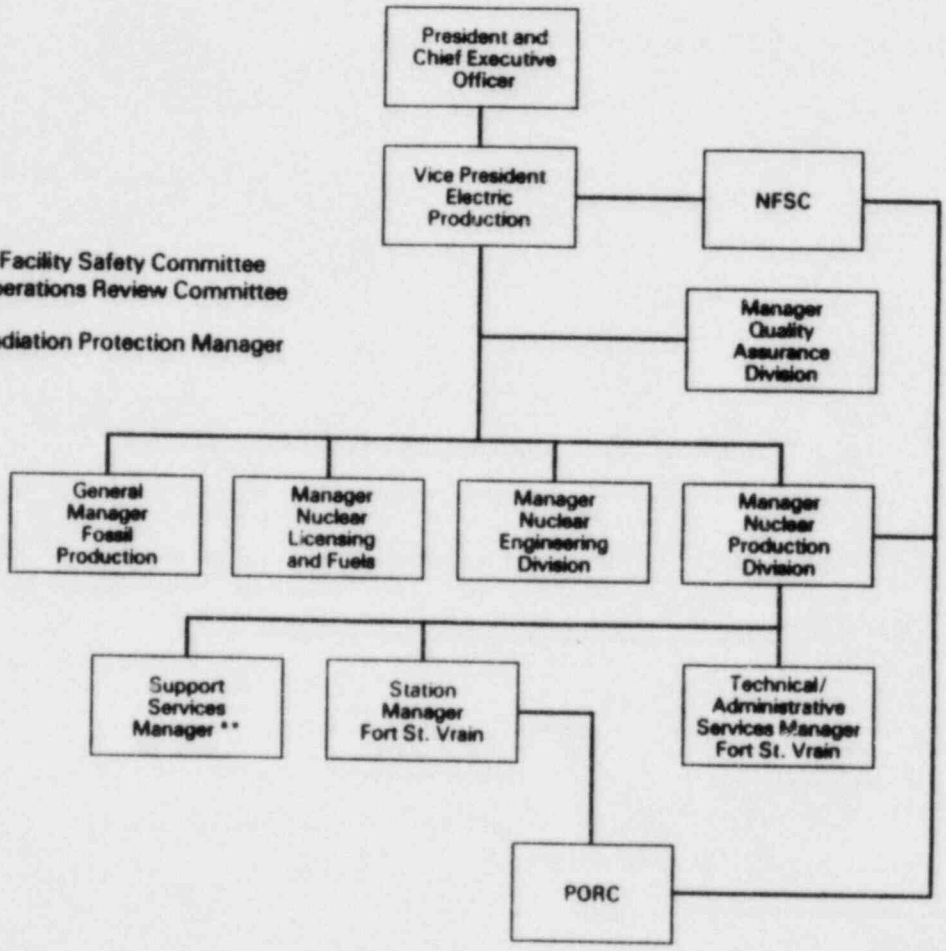
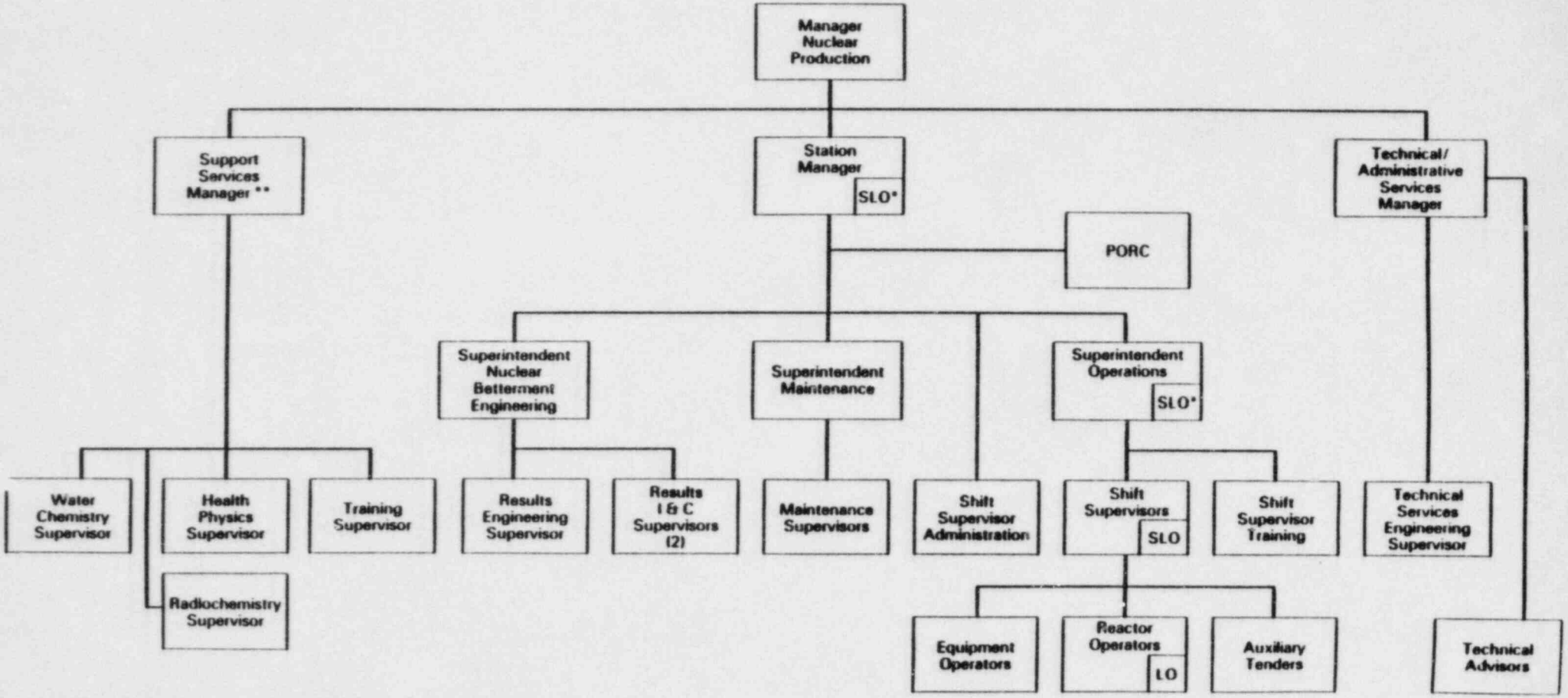


Figure 6.2-1

FORT ST. VRAIN NUCLEAR GENERATING STATION
CONDUCT OF OPERATIONS CHART



* Either the Station Manager or the Superintendent of Operations shall possess a Senior License

** Serves as the Radiation Protection Manager

PORC - Plant Operations Review Committee
SLO - Senior Licensed Operator
LO - Licensed Operator

Figure 6.2–2

DRAFT
APR 1 1985

DRAFT

APR 1 1985

6.2.3 TECHNICAL ADVISORS

FUNCTION

6.2.3.1 The Technical Advisors shall function to make objective evaluations of plant conditions and to advise or assist plant management in correcting conditions that may compromise safety of operations.

RESPONSIBILITIES

6.2.3.2 The Technical Advisors are responsible for:

a. Maximizing plant safety during and after accidents, transients, and emergencies by independently assessing plant conditions and by providing technical assistance to mitigate and minimize the effects of such incidents and make recommendations to the Superintendent of Operations,

b. Reviewing abnormal and emergency procedures,

c. Assisting the operations staff in applying the requirements of the Technical Specifications,

d. Providing evaluation of Licensee Event Reports from other plants as assigned, and

e. The Technical Advisor shall be in the Control Room within one hour after an emergency call. The Technical Advisors shall work on a normal day work schedule, but will be placed "on call" after normal working hours.

AUTHORITY

6.2.3.3 The Technical Advisors shall report to, and be directly responsible to, the Technical/Administrative Services Manager. The Technical Advisors shall maintain independence from normal plant operations to be objective in their evaluations.

DRAFT

APR 1 1985

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Upon commencement of commercial operation, the staffing of the plant shall be in accordance with American National Standards Institute N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants."

6.3.2 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable position, except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September, 1975.

6.3.3 Members of the plant staff who perform safety related functions (e.g., Senior Reactor Operators, Reactor Operators, Auxiliary Operators, Health Physics technicians, and key maintenance personnel) should, to the extent practicable, work an eight-hour day, 40-hour week, while the plant is operating. In the event that unforseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown or refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven-day period (all excluding shift turnover time).

c. A break of at least eight hours should be allowed between work periods (including shift turnover time).

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on shift.

6.3.4 If unusual circumstances arise requiring deviation from the above guidelines, such deviation shall be authorized by the Station Manager or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Station Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

Authorized deviations to the working hour guidelines shall be documented and available for Nuclear Regulatory Commission review.

6.4 Training

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10CFR Part 55. Compliance with Section 5.5 of ANSI 18.1-1971 shall be achieved no later than six months following commencement of commercial operation.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except for Fire Brigade training/drill sessions which shall be held at least once per calendar quarter.

6.4.3 The Technical Advisors shall have a bachelor degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents. An initial training and retraining program for the Technical Advisors shall be maintained under the direction of the Training Supervisor. The Technical Advisors shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the Control Room.

6.5 Review and Audit

6.5.1 Plant Operations Review Committee (PORC)

FUNCTION

6.5.1.1 The Plant Operations Review Committee shall function to advise the Station Manager on all matters related to nuclear safety.

DRAFT
APR 1 1985

COMPOSITION

6.5.1.2 The Plant Operations Review Committee shall be composed of the following:

Chairman: Station Manager
Technical/Administrative Services Manager
Support Services Manager
Superintendent of Operations
Superintendent of Maintenance
Superintendent of Nuclear Betterment Engineering
Health Physics Supervisor
Results Engineering Supervisor
Results Instrumentation and Controls Supervisor (2)
Shift Supervisor Administration
Shift Supervisor Training
Technical Services Engineering Supervisor
Training Supervisor

Alternates

6.5.1.3 An alternate chairman and alternate members, if required, shall be appointed in writing by the PORC Chairman to serve in the absence of a chairman or a member; however, no more than two alternate members shall participate in PORC activities at any one time.

Meeting Frequency

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the Chairman.

Quorum

6.5.1.5 A quorum shall consist of the Chairman or alternate Chairman, and four members including alternates.

Responsibilities

6.5.1.6 The PORC shall be responsible for:

a. Review of all procedures required by Technical Specification 6.8(1), (2), and (3) and changes thereto, and any other proposed procedure or changes to approved procedures as determined by the Station Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

DRAFT
APR 1 1985

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering the evaluation and recommendations to prevent recurrence to the Manager, Nuclear Production and to the Chairman of the Nuclear Facility Safety Committee.

f. Review of all Reportable Events.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations, and reports thereon as requested by the Chairman of the Nuclear Facility Safety Committee.

i. Review of the Plant Security Plan and implementing procedures and submittal of recommended changes to the Chairman of the Fort St. Vrain Security Committee.

j. Review of the plant Radiological Emergency Response Plan and implementing procedures.

k. Review of every unplanned onsite release of radioactive material to the environs, including the preparation of reports concerning evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Manager, Nuclear Production and the Nuclear Facility Safety Committee (NFSC).

Authority

6.5.1.7 The PORC shall:

a. Function to advise the Manager, Nuclear Production on all matters that affect nuclear safety.

b. Recommend to the Manager, Nuclear Production in writing, approval or disapproval of items considered under 6.5.1.6.a through 6.5.1.6.d above.

DRAFT
APR 1 1985

c. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6.a through 6.5.1.6.e above constitutes an unreviewed safety question.

d. Provide immediate written notification to the Manager, Nuclear Production and the Chairman of NFSC of disagreement between the PORC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.5.1.7.a above.

Records

6.5.1.8 The PORC shall maintain written minutes of each meeting and copies shall be provided to the Manager, Nuclear Production and Chairman of the Fort St. Vrain Nuclear Facility Safety Committee.

DRAFT

APR 1 1985

6.5.2 Nuclear Facility Safety Committee (NFSC)

Function

6.5.2.1 The Nuclear Facility Safety Committee shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear Power Plant Operations

b. Nuclear Engineering

c. Chemistry and Radiochemistry

d. Metallurgy

e. Instrumentation and Control

f. Radiological Safety

g. Mechanical and Electrical Engineering

h. Quality Assurance Practices

i. (Other appropriate fields associated with the unique characteristics of the nuclear power plant.)

COMPOSITION

6.5.2.2 The NFSC shall be composed of the following:

Chairman: Vice President, Electric Production
Manager, Nuclear Licensing and Fuels Division
Manager, Nuclear Production Division
Manager, Nuclear Engineering Division
Manager, Quality Assurance Division
Manager, Risk Management

Consultants, as required and appointed by the Chairman

ALTERNATES

6.5.3.3 An alternate chairman and alternate members, if required, shall be appointed in writing by the Chairman; however, no more than two alternate members shall participate in NFSC activities at any one time.

DRAFT
APR 1 1985

CONSULTANTS

6.5.3.4 Consultants shall be utilized as determined by the Chairman, NFSC, to provide expert advice to the NFSC.

MEETING FREQUENCY

6.5.3.5 The NFSC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.3.6 A quorum of the NFSC shall consist of the Chairman or his designated alternate and a majority of the NFSC members including alternates. No more than a minority of the quorum shall have line responsibilities for operation of the facility.

REVIEW

6.5.3.7 The Nuclear Facility Safety Committee shall review:

a. The safety evaluations for safety significant changes to procedures, equipment, or systems and safety significant tests or experiments completed under the provision of Section 50.59, 10CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in Section 50.59, 10CFR.

c. Proposed tests or experiments which involve an unreviewed safety question, as defined in Section 50.59, 10CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violation of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions affecting nuclear safety.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

DRAFT

APR 1 1985

g. All Reportable Events.

h. Any indication that there may be a deficiency in some aspect of design or operation of structures, systems, or components, that affect nuclear safety.

i. Reports and meeting minutes of the PORC.

AUDITS

6.5.3.8 Audits of facility activities shall be performed under the cognizance of the Nuclear Facility Safety Committee. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training, and qualifications, of the facility staff at least once per year.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or method of operation that affect nuclear safety at least once per six months.

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10CFR50, at least once per two years.

e. The review of the facility Emergency Plan and implementing procedures at least every 12 months by persons who have no direct responsibility for implementation of the Emergency Plan.

f. The review of the facility Security Plan and implementing procedures at least every 12 months by individuals independent of both security program management and personnel who have direct responsibility for implementation of the Security Plan.

g. Any other area of facility operation considered appropriate by the NFSC.

h. With respect to fire protection:

(i) an annual fire protection and loss prevention inspection and audit utilizing either qualified off-site licensee personnel or an outside fire protection firm;

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APR 1 1985

(ii) a biennial audit of the fire protection program and implementing procedures;

(iii) a triennial fire protection and loss prevention inspection and audit utilizing an outside qualified fire consultant.

i. The offsite Dose Calculation Manual and Process Control program and implementing procedures at least once per 24 months.

j. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months.

k. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21 Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975, at least once per 12 months.

AUTHORITY

6.5.3.9 The NFSC shall report to and advise the Vice President, Electric Production on those areas of responsibility specified in 6.5.3.7 and 6.5.3.8 above.

RECORDS

6.5.3.10 Records of NFSC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NFSC meeting shall be prepared, approved, and forwarded to the Vice President, Electric Production within 30 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.3.7.a, above shall be forwarded to the Vice President, Electric Production within 30 days following completion of the review.

c. Audit reports encompassed by Section 6.5.3.8, above shall be forwarded to the Vice President, Electric Production and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENTS ACTION

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE EVENT requiring 24 hour notification to the Commission shall be reviewed by the PORC and submitted to the NFSC and Station Manager.

DRAFT

APR 1 1900

6.7 SAFETY LIMIT VIOLATIONS

If a safety limit is exceeded, as defined in Specifications SL 2.1 and 2.2, the following actions shall be taken:

a. In accordance with 10CFR 50.72, the NRC Operations Center shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. The Manager, Nuclear Production and the NFSC shall be notified within 24 hours.

b. A Licensee Event Report shall be prepared in accordance with 10CFR 50.73.

c. The License Event Report shall be submitted to the Commission in pjaccordance with 10CFR 50.73, and to the PORC, the NFSC, and the Manager, Nuclear Production within 30 days after discovery of the event.

d. Critical operation of the unit shall not be resumed until authorized by the Nuclear Regulatory Commission.

DRAFT

APR 1 1985

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, November 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security plan implementation.

e. Emergency plan implementation.

f. Fire protection program implementation.

g. Process control program (PCP) implementation.

h. Offsite dose calculation manual (ODCM) implementation.

i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21 June 1974 and Regulatory Guide 4.1 Revision 1., April 1975.

j. Fuel surveillance program implementation.

k. Personnel radiation protection.

6.8.2 Procedures and administrative policies of a) above, and changes thereto, shall be reviewed by the PORC and approved by the appropriate Manager prior to implementation and reviewed periodically as set forth in Administrative Procedures.

Security Plant procedures, and changes thereto, shall be reviewed by the Plant Operations Review Committee and approved by the designated Plant Security Officer prior to implementation.

Security Plan procedures and changes thereto, shall be reviewed by the Fort St. Vrain Security Committee.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered,

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operators License, and

c. The change is documented, reviewed by the PORC and approved by the appropriate Superintendent within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in reactor building areas under accident condition. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

b. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions, and

 DRAFT

APR 1 1985

(vi) A procedure identifying (a) and authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

c. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and Reactor Building atmosphere samples under accident conditions. The program shall incude the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

DRAFT

APR 1 1985

6.9 REPORTING REQUIREMENTS

6.9.1 ROUTINE REPORTS AND REPORTABLE EVENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1 1a. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

b. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

c. Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

DRAFT

APR 1 1985

ANNUAL REPORTS

6.9.1.2 Annual reports covering the activities of the unit for the previous calendar year shall be submitted as described below. Reports required on an annual basis shall include:

a. Annual Occupations Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling shall be submitted to the Regional Administrater of the Nuclear Regulatory Commission Regional Office prior to March 31 of each year. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the whole body dose received from external sources shall be assigned to specific major work functions.

b. Annual Radiological Environmental Monitoring Report

A report on the Radiological Environmental Monitoring Program for the previous calendar year shall be submitted to the Regional Administrator of the Nuclear Regulatory Commission Regional Office (with a copy to the Director, Office of Nuclear Reactor Regulation) as a separate document by May 1 of each year.

The Annual Radiological Environmental Monitoring Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental monitoring activities for the report period, including a comparison with pre-operational studies, operational controls (as appropriate), and previous environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land-use censuses required by Specification ELCO 8.2.1. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analyses of the problem and a planned course of action to alleviate the problem.

The Annual Radiological Environmental Monitoring Reports shall include the results of analysis of all radiological environmental samples and of all measurements taken during the period pursuant to the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November, 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: A summary description of the Radiological Environmental Monitoring Program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the center line of the reactor; the results of licensee participation in the interlaboratory comparison program; and discussion of all analyses in which the lower limits of detection required by Table 8.2-2 was not achievable.

* One map shall cover stations near SITE BOUNDARY; a second shall include the more distant stations.

DRAFT

APR 1 1985

If the Radiological Environmental Monitoring Program is not being conducted as specified in Table 8.2-1, in lieu of a Licensee Event Report, a description of the reasons the program was not conducted as required and the plan for preventing recurrence shall be prepared and submitted to the Nuclear Regulatory Commission in the Annual Radiological Environmental Monitoring Report.

c. Annual Diesel Generator Reliability Report

An annual data report on diesel generator reliability shall be submitted to the NRC in a manner consistent with Regulatory Guide 1.108 position C.3.a prior to March 1 of each year.

d. Annual Meteorological Data Report

An annual summary of hourly meteorological data collected over the previous year shall be maintained for five years by the licensee. This annual summary may be either in the form of an hour-by hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

DRAFT

APR 1 1985

SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.3 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous six months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June, 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. The summary report, submitted within 60 days of January 1 of each year, shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year, and shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 5.3-1) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time, and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. (Conservative approximate methods are acceptable). The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Revision 1, October, 1977.

 DRAFT

APR 1 1985

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, large quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

DRAF
APR 1 19

MONTHLY OPERATING REPORT

6.9.1.4 A routine operating report covering the operation of the unit during the previous month shall be submitted prior to the fifteenth calendar day of the following month. Submittal shall be to the Director, Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, ATTN: Document Control Desk with a copy to the appropriate NRC Regional Administrator.

Each monthly operating report shall include:

a. A narrative summary of operating experience during the report period relating to safe operation of the facility, including major safety-related maintenance.

b. Report of any single release of radioactivity or radiation exposure which accounts for more than 10% of the allowable annual values.

c. Indications of failed fuel resulting from irradiated fuel examinations, completed during the report period.

d. The monthly statistical information contained in Regulatory Guide 1.16.

REPORTABLE EVENTS

6.9.1.5 The REPORTABLE EVENTS of Specifications 6.9.1.6 and 6.9.1.7 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of the event. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.6 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Administrator of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 30 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a Limiting Condition for Operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel coatings, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% delta k/k; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% delta k/k; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the FSAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the FSAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

 DRAFT

APR 1 1985

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action of corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specifications 8.1.1.h or 8.1.2.g.

k. Exceeding the limits in Specification 8.1.1.c for the maximum amount of gaseous radioactivity in a gas waste surge tank. The written followup report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

THIRTY DAY WRITTEN REPORTS

6.9.1.7 The types of events listed below shall be the subject of written reports to the Administrator of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in

reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.6c above designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of:

1. More than 1 curie of radioactive material in liquid effluents,

2. More than 150 curies of nobel gas in gaseous effluents, or

3. More than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information.

1. A description of the event and equipment involved

2. Cause(s) for the unplanned release,

3. Actions taken to prevent recurrence, and

4. Consequences of the unplanned release.

f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 8.2-3 when averaged over any clendar quarter sampling period.

g. Exceeding the limits in Specification 8.1.5.a, for the maximum radioactive effluent releases. The written followup report shall define the corrective action to be taken to reduce subsequent releases and prevent recurrence of exceeding the above limits. It shall also include a schedule for achieving conformance with the above limits.

SPECIAL REPORTS

6.9.2 Special Reports shall be submitted to the NRC Regional Administrator within the time period specified for each report.

DRAFT
APR 1 1985

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulation, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All reportable events submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source leak tests and results.

h. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Fort St. Vrain #1 Operating License.

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

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

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the NFSC.

l. Records of the service lives of all snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Records of reactor tests and experiments.

o. Records for Environmental Qualification which are covered under provisions of 10 CFR 50.49 "Equipment Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants".

p. Records and logs pertaining to the environmental monitoring program including baseline data from the pre-operational environmental monitoring programs (both radiological and non-radiological).

DRAFT
APR 1 1985

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

Respiratory protective equipment shall be provided in accordance with 10 CFR 20.103.

DRAFT
APR 1 1985

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each area accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 100 mrem but less than 1000 mrem shall be barricaded and conspicuously posted-as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individual permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physics staff in the RWP.

6.12.2 In addition to the requirements of 7.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked enclosures to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision. Enclosures shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area.

DRAFT
APR 1 1985

For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

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* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

**Measurement made at 18" from source of radioactivity.

DRAFT
APR 1 1985

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

DRAFT
APR 1 1985

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

DRAFT
APR 1 1985

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS

6.15.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto:

5. An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC, in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto:

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

DRAFT
APR 1 1985

6.16 FUEL SURVEILLANCE PROGRAM

6.16.1 The Fuel Surveillance program shall be approved by the Commission prior to implementation. Any changes to the program require Commission approval and concurrence prior to implementation.

6.16.2 The Fuel Surveillance Program shall include:

(i) Procedures for obtaining a photographic record of all six vertical faces of at least 90% of the spent fuel elements removed from the core during each refueling using the Fuel Handling Machine 35mm camera or the Cask Video Monitor.

(ii) Procedures for evaluating all photographic records for indications of significant abnormalities which could have an effect on the structural integrity of the elements prior to returning to power operation following a refuelling.

(iii) Upon discovery of any significant abnormalities which could have an effect on the structural integrity of a fuel element, provisions for:

a. Informing the Commission within three days.

b. Performing an engineering evaluation to determine the cause of the abnormality.

c. Receiving Commission approval prior to resuming reactor operation.

iv) Provisions to submit all aspects of the examinations performed on discharged fuel elements to the Commission in a timely fashion.

DRAFT

APR 1 1985

The RADIOLOGICAL and ENVIRONMENTAL Technical Specifications (Chapter 8.0 of the existing Technical Specifications) are not included in this submittal. Section 8.0 is not being revised under the present scope of the Technical Specification Upgrade Program.

ENCLOSURE 2

The draft Technical Specifications have been significantly reorganized. Many of the existing specifications have been either relocated, combined with others, or deleted. All Technical Specification deletions went through a rigorous review cycle to ensure that the deletion was appropriate. The attached Tech Spec Position Papers justify the deletion of the Breathing Air System, the Core Differential Pressure Indicator, and the Fuel Loading and Rise-To-Power Specifications. To facilitate your review, Table 2-1 lists existing FSV Technical Specifications that have been combined with others.

TABLE 2-1
RELOCATED TECHNICAL SPECIFICATIONS

Existing FSV Specification	Title	Relocation	Explanatory Footnote
4.1.4	Partially Inserted Control Rods	3.1.4	(1)
4.2.5/5.2.24	Circulating Water Makeup System	3/4.5.4	(1)
4.2.12	Liquid Nitrogen Storage	3.7.5	(1)
4.3.5	Storage Ponds	3.5.4	(1)
4.4.6/5.4.13	Room Temperature, 480V Switchgear	3/4.3.3	(1)
4.5.2	Reactor Vessel Internal Maintenance	3.9.1	(1)
4.9.2	Dew Point Moisture Monitors	3.3.1-Footnote (k)	(2)
4.10.1/5.10.1	Three Room Control Complex Isolation Dampers	3/4.7.6.3	(1)
5.2.6	Plateout Probe	4.4.1.e	(1)
5.2.13	PCRV Concrete Helium Permeability	4.6.4.5.f	(1)
5.2.14	PCRV Liner Corrosion	4.6.4	(1)
5.3.2	Main Hot and Reheat Steam Stop Check Valves	4.5.2.1.a	(3)
5.3.4	Safe Shutdown Cooling Valves	Various	(4)
5.3.9	S.G. Safety Valves	3.5.2.1	(1)
5.3.10	Secondary Coolant System Instrumentation	4.5.2.1.a	(5)
5.4.3	Core Region Outlet Temperature	4.2.2	(1)
5.10.3	Smoke Detectors and Alarm	4.3.2.5	(1)
3.0, 4.1.0, 4.2.0, 4.3.0, 4.4.0, 4.5.0, 4.6.0, 4.7.0, 4.9.0, 4.10.0, 5.1.0, 5.2.0, 5.3.0, 5.4.0, 5.5.0, 5.6.0, 5.7.0, 5.10.0	Objective and Applicability Sections	Bases Sections	(6)
5.2.2.a & b	PCRV Tendon Corrosion (Sample Wire and Atmosphere Inspection)	4.6.4	(7)

FOOTNOTES FOR TABLE 2-1

(1) The requirements of existing FSV Technical Specification have been directly relocated into the proposed cross-referenced Specification.

(2) LCO 4.9.2 was initiated as a "one-time" test of the Plant Protection System Dewpoint Moisture Monitors at various power levels of 5%, 25%, and 100%. Successful completion of this test was performed at power levels of 5%, 25%, and 70%. The data collected during these tests was extrapolated up to a power level of 100% with favorable results. Rise to power testing began in April, 1975 and was completed in May, 1978 through and including 70% power. Therefore, this LCO is eliminated. However, to retain operator awareness and LCO status of any future PPS moisture monitor tests, Specification 3.3.1, Table 3.3.1-1 footnote (k), has been revised to reflect the permissible bypass conditions.

(3) SR 5.3.2 requires partial and full stroke testing of the Main and Hot Reheat Steam Stop Check Valves. The specific requirement for performing individual surveillances on the Main and Hot Reheat Steam Stop Check Valves will be retained and included in surveillance procedures. However, Surveillance Requirement 4.5.2.1.a will require demonstration of an OPERABLE safe shutdown cooling flow path through the steam generator heat sink and will trigger the requirement to stroke these check valves. Consequently, the requirement to maintain OPERABLE Main and Reheat Steam Stop Check Valves is retained under the definition for OPERABILITY of the steam generators.

(4) SR 5.3.4 requires testing of the Safe Shutdown Cooling Valves. Verification of valve operability is ensured by normal operation or they will be demonstrated operable by the Surveillance Requirements on Safe Shutdown Cooling systems. The Safe Shutdown Cooling system surveillances will typically require valve alignment checks, pump functional tests, and/or simulation of safe shutdown cooling conditions (e.g. demonstration of flow to Steam Generator sections). The surveillance procedures associated with these Surveillance Requirements will specifically ensure that the Safe Shutdown Cooling Valves are tested.

(5) SR 5.3.10 requires a functional test and calibration of the secondary coolant reheat steam instrumentation. Verification of the instrumentation operability will be ensured during the safe shutdown cooling flow test required by Surveillance Rquirement 4.5.2.1.a. The specific requirement for calibrating the instrument will be included in the surveillance procedures.

(6) Pursuant to 10CFR 50.36 and ANSI/ANS-58.4-1979, the Bases sections should include the summary statements of the reasons for the Technical Specifications.

FOOTNOTES FOR TABLE 2-1

(7) The PCRV Tendon Corrosion and Anchor Assembly Surveillance has been upgraded to include a more complete and thorough inspection and evaluation program as described in PSC Letter D. Warembourg to E. Johnson dated March 5, 1985 (P-85071).

TECH SPEC POSITION PAPER - DELETION OF FUEL LOADING AND INITIAL RISE TO POWER LCO's - 4.9.1

TITLE: Fuel Loading and Initial Rise to Power

LCO 4.9.1 addressed two phases of the initial power ascension test program. Phase 1 included fuel loading and low power physics testing in an air or helium environment. Phase 2 included hot physics tests with helium environment and rise to full power testing.

These low power physics tests began in January, 1974 and were completed in April, 1975. These tests were a prerequisite for the rise-to-power tests which began in April, 1975.

Based on the original Technical Specification requirement that this test be a "one-time" initial low power physics test and having satisfactorily completed the testing in April, 1975, LCO 4.9.1 is eliminated. REFERENCE: PSC Letter: J. W. Gahm to E. H. Johnson dated March 5, 1985 (P-85063).

TECHNICAL SPECIFICATION POSITION PAPER - DELETION OF SR 5.4.6

Existing Specification

Specification SR 5.4.6 - Core Delta P Indicator - Surveillance Requirement

The core Delta P instrumentation shall be calibrated on a once per refueling cycle interval.

Justification for Deletion

The core delta pressure indicator (PDT-1112) was used to monitor core pressure drop during rise-to-power tests and fluctuation testing. This pressure indicator is not part of the plant protective system, the plant's regulating system, nor is it part of the plant's nuclear instrumentation.

In accordance with 10 CFR 50.36, ANSI/ANS-58.4-1979, and the Work Specification for the Technical Specification Upgrade Program (WS-TS-1) the Technical Specifications are derived from the evaluations and analyses included in the FSAR. Furthermore, the Technical Specifications shall contain only those items relied upon in the safety analyses and/or those items specifically required by Federal Regulations.

The core delta pressure indicator is not relied upon in the Fort St. Vrain safety analyses/evaluations. Credit is taken for the instrument in FSAR Subsection 7.3.3.2 as an indicator of the total core pressure drop. However, the indicator is not used in response to any of the accident analyses. During normal operation, fluctuations in core differential pressure are indicated by the region outlet temperature thermocouples (Specification 3/4.2.2).

The FSAR, 10CFR 50.36, 10CFR 50 Appendix A, Reg Guide 1.97 and other associated documents were reviewed to determine if an LCO and an Associated Surveillance Requirement (SR) are necessary for the core delta pressure indicator. Neither of these documents nor the Standard Technical Specifications require a condition or limitation upon reactor operation with respect to an inoperable core differential pressure indicator.

In summary, the core Delta P indicator is not relied upon in the Fort St. Vrain safety analyses nor is it required by Federal Regulations to be included in the Technical Specifications. Furthermore, the indicator does not fall under the "immediate threat" standard (1). Consequently, the Surveillance Requirement SR 5.4.6 will be deleted from the Technical Specifications and will be appropriately included within the plant's Administrative Controls Program.

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(1) The Atomic Safety and Licensing Appeal Board has propagated an "immediate threat" standard for defining what should be included in the Technical Specifications. In ALAB-531, the Board stated that: "___ as best we can discern it, the contemplation of both the act and the regulations is that Technical Specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." (In the Matter of Portland General Electric Company, et al. (Trojan Nuclear Power Plant), 9 NRC 263 (1979).)

TECHNICAL SPECIFICATION POSITION PAPER-DELETION OF S.R. 5.10.5

Existing Specification:

Specification SR 5.10.5-Breathing Air System, Surveillance Requirement

The operability of the Breathing Air System shall be demonstrated annually, as follows:

a. Functionally test the compressors and air supply piping.

b. Test the quality of the air supplied.

Justification for Deletion:

The breathable air system is designed to provide a continuous supply of purified air to air-hose line type respirators for use in the Control Room, Auxiliary Electric Equipment Room, and the 480V Switchgear Room. The system is intended to allow the Fire Brigade personnel to enter these areas for firefighting activities. The system can also be used by operating personnel for reactor operation activities during conditions when the room air could be potentially dangerous to health.

In accordance with 10CFR 50.36, ANSI/ANS-58.4-1979, and the work specification for the Technical Specification Upgrade Program (WS-TS-1) the Technical Specifications are derived from the evaluations and analyses included in the FSAR. Furthermore, the Technical Specifications shall contain only those items relied upon in the safety analyses and/or those items specifically required by Federal Regulations.

The breathable air system is not relied upon in the Fort St. Vrain safety analyses/evaluations. Credit is taken for the system as a supplement to the self-contained breathing appartus system. Furthermore, neither of the breathing air systems will be necessary should the plant's fixed suppression systems and associated HVAC systems be OPERABLE.

The FSAR, 10CFR 50.36 App. R, Reg Guide 1.120, and other associated documents were reviewed to determine if an LCO and an Associated Surveillance Requirement (SR) are necessary for the plant's Breathing Air System. Neither of these documents nor the Standard Technical Specifications require a condition or limitation upon reactor operation with respect to an inoperable Breathing Air System.

10CFR50 App. R does specify that surveillance procedures be established to ensure that fire barriers are in place and that

fire suppression systems are operable. It does not specify that the breathable air system and other Fire Brigade protective equipment (i.e., hard hats, emergency communications equipment, portable extinguishers, etc.) be included in the Technical Specifications.

In summary, the breathable air system is not relied upon in the Fort St. Vrain safety analyses nor is it required by Federal regulations to be included in the Technical Specifications. Furthermore, the breathable air system does not fall under the "immediate threat" standard (1). Consequently, the surveillance requirement on the breathing air system will be deleted from the Technical Specifications and will be appropriately included within the plant's Administrative Controls Program.

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(1) The Atomic Safety and Licensing Appeal Board has propagated an "immediate threat" standard for defining what should be included in the Technical Specifications. In ALAB-531, the Board stated that: ".... as best we can discern it, the contemplation of both the act and the regulations is that Technical Specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." (In the Matter of Portland General Electric Company, et al. (Trojan Nuclear Power Plant), 9 NRC 263 (1979).)

Enclosure 3

FORT ST. VRAIN NUCLEAR GENERATING STATION

UPGRADE TECHNICAL SPECIFICATIONS

CROSS REFERENCE TO EXISTING TECHNICAL SPECIFICATIONS

SECTION	DESCRIPTION	EXISTING FSV T.S. SECTION (LCO/SR)
	INTRODUCTION	1.0
1.0	DEFINITIONS	2.0
2.0	SAFETY LIMITS and LIMITING SAFETY SYSTEM SETTINGS	
2.1	Safety Limits	
2.1.1	Reactor Core	3.1
2.1.2	Reactor Vessel Pressure	3.2
2.2	Limiting Safety System Settings	3.3
2.2.1	Trip Setpoints	
	LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS	
3/4.0	APPLICABILITY	4.0/5.0
3/4.1	REACTIVITY CONTROL SYSTEMS	
3/4.1.1	Control Rods	
3/4.1.1.1	Operable Control Rods/ Shutdown Margin-Operation	4.1.2/5.1.1
3/4.1.1.2	Operable Control Rods/ Shutdown Margin-Shutdown	4.1.2/5.1.1
3/4.1.2	Control Rod Position Instrumentation	
3/4.1.2.1	Fully Inserted and Fully Withdrawn Rod Pair	New/5.1.1
3/4.1.2.2	Partially Inserted Rod Pair	New/5.1.1
3/4.1.3	Control Rod Penetration Purge Flow	New/5.1.1
3/4.1.4	Control Rod Sequence and Position Requirements	4.1.3, 4.1.4/5.1.5
3/4.1.5	Reactivity Change with Temperature	4.1.5/5.1.3

Enclosure 3

FORT ST. VRAIN NUCLEAR GENERATING STATION

UPGRADE TECHNICAL SPECIFICATIONS

CROSS REFERENCE TO EXISTING TECHNICAL SPECIFICATIONS

3/4.1.6	Reserve Shutdown System	4.1.6/5.1.2
3/4.1.7	Reactivity Status	4.1.8/5.1.4
3/4.2	CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS	
3/4.2.1	Core Irradiation	4.1.1/5.1.8
3/4.2.2	Core Inlet Orifice Valves/ Region Outlet Temperature Limits	4.1.7/5.1.7
3/4.2.3	Core Inlet Orifice Valves/ Comparison Regions	4.1.7/5.1.7
3/4.2.4	Core Inlet Orifice Valves/ Minimum Helium Flow	4.1.9/5.1.9
3/4.2.5	Region Constraint Devices	New/5.2.26
3/4.3	INSTRUMENTATION	
3/4.3.1	Plant Protective System	4.4.1/5.4.1
3/4.3.2	Monitoring Instrumentation	
3/4.3.2.1	Analytical Moisture Monitors	4.4.5/5.4.12
3/4.3.2.2	Radiation Monitoring Instrumentation	4.4.3/5.4.9
3/4.3.2.3	Seismic Instrumentation	4.4.4/5.4.10
3/4.3.2.4	Meteorological Instrumentation	New/5.4.16
3/4.3.2.5	Fire Detection and Alarm System	4.10.3/5.4.2
3/4.3.2.6	Chlorine Detection and Alarm System	New/5.4.14
3/4.3.2.7	Power-to-Flow Ratio Recording Instrumentation	New/5.4.8
3/4.3.3	Three Room Control Complex Temperature Monitoring	4.4.2/5.4.7

Enclosure 3

FORT ST. VRAIN NUCLEAR GENERATING STATION

UPGRADE TECHNICAL SPECIFICATIONS

CROSS REFERENCE TO EXISTING TECHNICAL SPECIFICATIONS

3/4.4	PRIMARY COOLANT	
3/4.4.1	Primary Coolant Activity	4.2.8/5.2.11
3/4.4.2	Loop Impurity Levels-High Temperature	4.2.10/5.2.12, 5.2.22, 5.2.25
3/4.4.3	Loop Impurity Levels-Low Temperature	4.2.11/5.2.12
3/4.5	SAFE SHUTDOWN COOLING SYSTEMS	
3/4.5.1	Helium Circulators	4.2.1, 4.2.2/5.2.7, 5.2.8, 5.2.9, 5.2.23
3/4.5.1.1	Helium Circulators - Power Operation and Low Power	5.2.27
3/4.5.1.2	Helium Circulators - Startup, Shutdown and Refueling	
3/4.5.2	Steam Generators	4.3.1/5.2.7, 5.3.1, 5.3.10, 5.3.11, 5.3.12
3/4.5.2.1	Steam Generators - Power Operation and Low Power	
3/4.5.2.2	Steam Generators - Startup	
3/4.5.2.3	Steam Generators - Shutdown and Refueling	
3/4.5.3	Emergency Condensate and Emergency Feedwater Headers	4.3.4/5.2.7, 5.3.1
3/4.5.3.1	Emergency Condensate and Emergency Feedwater Headers - Power Operation and Low Power	
3/4.5.3.2	Emergency Condensate and Emergency Feedwater Headers - Startup	
3/4.5.3.3	Emergency Condensate and Emergency Feedwater Headers - Startup and Refueling	
3/4.5.4	Firewater Supply System	4.2.6/5.2.10

Enclosure 3

FORT ST. VRAIN NUCLEAR GENERATING STATION

UPGRADE TECHNICAL SPECIFICATIONS

CROSS REFERENCE TO EXISTING TECHNICAL SPECIFICATIONS

3/4.6	PCRV AND CONFINEMENT SYSTEMS	
3/4.6.1	PCRV Pressurization	
3/4.6.1.1	PCRV Safety Valves	4.2.7/5.2.1
3/4.6.1.2	Steam Generator/ Circulator Penetrations	4.2.7/5.2.1
3/4.6.1.3	Interspace Pressurization	4.2.7/5.2.15
3/4.6.1.4	Interspace Leakage	4.2.9/5.2.16
3/4.6.1.5	Interspace Radiation Monitoring	New -
3/4.6.2	PCRV Liner Cooling System	4.2.13, 4.2.14/5.4.4 5.4.11
3/4.6.3	PCRV Liner Cooling System Temperatures	4.2.15/5.4.5
3/4.6.4	PCRV Integrity	New/5.2.2, 5.2.3,5.2.15
3/4.6.5	Reactor Building Confinement	
3/4.6.5.1	Reactor Building Integrity	4.5.1/5.5.1
3/4.6.5.2	Reactor Building Exhaust System	4.5.1/5.5.3
3/4.6.5.3	Reactor Building Over-Pressure Protection System	4.5.1/5.5.2
3/4.7	PLANT and SAFE SHUTDOWN COOLING SUPPORT SYSTEMS	
3/4.7.1	Turbine Cycle	
3/4.7.1.1	Boiler Feed Pumps	4.3.2/5.3.1, 5.2.7
3/4.7.1.2	Steam/Water Dump System	4.3.3/5.3.1
3/4.7.1.3	Pressure Relief Valves	4.3.13/5.3.3
3/4.7.1.4	Secondary Coolant Activity	4.3.8/5.3.7
3/4.7.2	Hydraulic Power System	4.3.7/5.3.5
3/4.7.3	Instrument Air System	4.3.6/5.3.6
3/4.7.4	Service Water System	4.2.4/5.6.1
3/4.7.5	Primary Coolant Depressurization	4.2.18/5.2.29
3/4.7.6	Fire Suppression Systems	
3/4.7.6.1	Spray and/or Sprinkler System	4.10.5/5.10.6
3/4.7.6.2	Carbon Dioxide Systems	4.10.6/5.10.7
3/4.7.6.3	Halon Systems	4.10.2/5.10.2
3/4.7.6.4	Fire Hose Stations	4.10.7/5.10.8
3/4.7.6.5	Yard Fire Hydrants and Hydrant Hose Houses	4.10.8/5.10.9

Enclosure 3

FORT ST. VRAIN NUCLEAR GENERATING STATION

UPGRADE TECHNICAL SPECIFICATIONS

CROSS REFERENCE TO EXISTING TECHNICAL SPECIFICATIONS

3/4.7.7	Fire Rated Assemblies	4.10.4/5.10.4
3/4.7.8	ACM Diesel Generator	4.2.17/5.2.20
3/4.7.9	Control Room Emergency Ventilation System	new/5.4.15
3/4.7.10	Snubbers	4.3.10/5.3.8
3/4.8	AUXILIARY ELECTRIC POWER SYSTEMS	
3/4.8.1	AC Power Sources	
3/4.8.1.1	AC Power Sources - Operating	4.6.1/5.6.1
3/4.8.1.2	AC Power Sources - Shutdown	4.6.1/5.6.1
3/4.8.2	DC Power Sources	
3/4.8.2.1	DC Power Sources - Operating	New/5.6.2
3/4.8.2.2	DC Power Sources - Shutdown	New/5.6.2
3/4.8.3	Onsite Power Distribution	
3/4.8.3.1	Onsite Power Distribution-Operating	New/5.6.3
3/4.8.3.2	Onsite Power Distribution-Shutdown	New/5.6.3
3/4.9	FUEL HANDLING and STORAGE SYSTEMS	
3/4.9.1	Fuel Handling and Maintenance In the Reactor	4.7.1/5.7.3
3/4.9.2	Fuel Handling Machine	4.7.2/5.7.1
3/4.9.3	Fuel Storage Well	4.7.3/5.7.2
3/4.9.4	Spent Fuel Shipping Cask	4.7.4/5.7.3
3/4.9.5	Communications During Core Alterations	New/5.7.5
3/4.10	SPECIAL TEST EXCEPTIONS	
3/4.10.1	Xenon Stability	

Enclosure 3

FORT ST. VRAIN NUCLEAR GENERATING STATION

UPGRADE TECHNICAL SPECIFICATIONS

CROSS REFERENCE TO EXISTING TECHNICAL SPECIFICATIONS

5.0	DESIGN FEATURES	6.0
5.1	Site	6.3
5.2	Reactor Coolant System and Steam Plant System	
5.2.1	Prestressed Concrete Reactor Vessel (PCRV)	6.2.1
5.2.2	Steam Generator Orifices	6.2.2
5.2.3	Steam Safety Valves	6.2.3
5.3	Reactor Core	6.1
5.3.1	Reactor Assembly	
5.3.2	Active Core	
5.3.3	Fuel	
5.3.4	Reload Segment Design	
5.3.5	Reflector	
5.4	Fuel Storage	
5.4.1	Criticality	
5.4.2	Containment	
5.4.3	Neutron Absorber Heat Sink	
5.5	Metorological Tower Location	
5.6	Component and Transient Cyclic Limit	
6.0	ADMINISTRATIVE CONTROLS	7.0
6.1	Responsibility	7.1.1
6.2	Organization	7.1.1
6.3	Unit Staff Qualifications	7.1.1
6.4	Training	7.1.1
6.5	Review and Audit	7.1.2, 7.1.3
6.5.1	Plant Operations Review Committee (PORC)	7.1.2
6.5.2	Nuclear Facility Safety Committee (NFSC)	7.1.3
6.6	Reportable Event Action	7.5.2
6.7	Safety Limit Violations	7.2
6.8	Procedures and Programs	7.4

Enclosure 3

FORT ST. VRAIN NUCLEAR GENERATING STATION

UPGRADE TECHNICAL SPECIFICATIONS

CROSS REFERENCE TO EXISTING TECHNICAL SPECIFICATIONS

6.9	Reporting Requirements	7.5
6.9.1	Routine Reports and Reportable Events	7.5.1
6.9.2	Special Reports	New
6.10	Record Retention	7.3
6.11	Radiation Protection Program	7.4
6.12	High Radiation Area	New
6.13	Process Control Program (PCP)	New
6.14	Offsite Dose Calculation Manual (ODCM)	New
6.15	Major Changes to Liquid, Gaseous and Solid Radwaste Treatment Systems	7.5
6.16	Fuel Surveillance Section	New
7.0	NOT USED	
8.0	RADIOLOGICAL and ENVIRONMENTAL	8.0