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May 27, 1988
Fort St. Vrain
Unit No. 1
P-88184

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Attention: Mr. Jose A. Calvo
Director, Project Directorate IV

Docket No. 50-267

SUBJECT: Technical Specification Upgrade Program, Revised Final Draft Submittal

REFERENCE: See Attached Reference List

Dear Mr. Calvo:

Attached is the revised Final Draft of the upgraded Fort St. Vrain (FSV) Technical Specifications. These specifications are submitted for your review as a Proof and Review draft.

The initial Final Draft (Reference 1) has been revised to address NRC comments provided in References 2 through 5. These comments were discussed in several meetings as summarized in References 6 through 11. Also, the Public Service Company of Colorado (PSC) resolutions of these comments have been incorporated, as documented in References 12 through 18.

Attachment 1 is the revised Final Draft Upgraded Technical Specifications. These specifications have been reviewed by a special subcommittee of the FSV PORC and their comments have been incorporated.

Attachment 2 is a discussion for each revision to technical specification drafts that were previously transmitted to the NRC. For each specification, the last docketed submittal is identified, all changes are listed, and a justification for each change is provided.

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This submittal is provided with the following additional considerations:

1. PSC's justifications for changes from the current FSV Technical Specifications and other "C" category comments will be provided via separate correspondence, by June 3, 1988.

2. The Administrative Controls, Section 6.0, has been revised to delete organization charts, under the guidance of Generic Letter 88-06. Several discussions within this Section still address organizational information, such as position titles. PSC has recently announced a re-organization that affects this Section and an update will be submitted as soon as practical.

3. PCRV Liner Cooling System tube temperature limits have been included but they are a significant restriction over the current FSV Technical Specifications. PSC is attempting to operate per these new limits and any concerns identified during this experience will be addressed as soon as practical.

PSC considers these revised Final Draft Upgraded Technical Specifications responsive to the goals of improving the clarity, completeness, and the correctness of the Technical Specifications, consistent with the FSV licensing basis as embodied in the FSAR. Furthermore, this submittal is supportive of the goal of implementation by the startup following the fourth refueling.

If you have any questions regarding this submittal, please contact Mr. M. H. Holmes at (303) 480-6960.

Very truly yours,

Lawrence Brey

H. L. Brey, Manager
Nuclear Licensing and
Resource Management

HLB/SWC/lmb

Attachments

cc: Regional Administrator, Region IV
ATTN: Mr. T. F. Westerman, Chief
Projects Section B

Mr. Robert Farrell
Senior Resident Inspector
Fort St. Vrain

Reference List:

1. PSC letter, Lee to Berkow, dated 11/27/85 (P-85448)
2. NRC letter, Heitner to Walker, dated 5/30/86 (G-86285)
3. NRC letter, Heitner to Williams, dated 4/17/87 (G-87131)
4. NRC letter, Heitner to Williams, dated 5/6/87 (G-87161)
5. NRC letter, Heitner to Williams, dated 7/2/87 (G-87217)
6. NRC memorandum, Hinson to Berkow, dated 10/28/86 (G-86566)
7. NRC memorandum, Hinson to Berkow, dated 12/15/86 (G-86635)
8. NRC letter, Heitner to Williams, dated 2/18/87 (G-87056)
9. NRC memorandum, Heitner to Calvo, dated 10/1/87 (G-87348)
10. NRC memorandum, Heitner to Calvo, dated 1/12/88 (G-88009)
11. NRC memorandum, Heitner to Calvo, dated 4/4/88 (G-88164)
12. PSC letter, Brey to Berkow, dated 2/20/87 (P-87063)
13. PSC letter, Brey to Calvo, dated 8/24/87 (P-87272)
14. PSC letter, Brey to Calvo, dated 11/19/87 (P-87410)
15. PSC letter, Brey to Calvo, dated 12/23/87 (P-87441)
16. PSC letter, Brey to Calvo, dated 2/2/88 (P-88045)
17. PSC letter, Brey to Calvo, dated 3/22/88 (P-88062)
18. PSC letter, Brey to Calvo, dated 3/8/88 (P-88082)

ATTACHMENT 1

to

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FORT ST. VRAIN NUCLEAR GENERATING STATION

UPGRADE TECHNICAL SPECIFICATIONS

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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 may not be changed except as part of a license amendment approved by the Nuclear Regulatory Commission.

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SECTION 1.0

DEFINITIONS

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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, considering system design, in conjunction with each possible interlock logic state and verification of the required logic output.

ALLOWABLE VALUE

1.3 The ALLOWABLE VALUE shall be the least conservative acceptable "as found" value for a TRIP SETPOINT.

BASES (BASIS)

1.4 The BASES shall summarize the reasons for the SAFETY LIMIT, the LIMITING SAFETY SYSTEM SETTINGS, the Limiting Condition of Operation, and the Surveillance Requirements. In accordance with 10 CFR 50.36, the BASES are not a part of the Technical Specifications.

CALCULATED BULK CORE TEMPERATURE (CBCT)

1.5 The CALCULATED BULK CORE TEMPERATURE (CBCT) shall be the calculated average temperature of the core, including graphite and fuel but not the reflector, assuming a loss of all forced circulation of PRIMARY COOLANT FLOW. Use of the CALCULATED BULK CORE TEMPERATURE is explained in Specification 3.0.5.

CHANNEL CALIBRATION

1.6 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, considering system design, 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.

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CHANNEL CHECK

1.7 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

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

COMPARISON REGION

1.9 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. To compensate for these discrepancies Regions 20 and 32 through 37 shall have their region outlet temperatures determined by the relative power and flow characteristics of other regions in the core referred to as COMPARISON REGIONS. These discrepancies are caused by a transverse flow of relatively cool helium from the core reflector interface along the region outlet thermocouple sleeves. This flow passes over the region outlet thermocouple assemblies of these regions and depresses the indicated outlet temperatures.

CONGESTED CABLE AREAS(S)

1.10 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

1.11 CORE ALTERATION shall be the movement or manipulation of any component within the PCRV that alters the core reactivity except for insertion of control rod pairs or reserve shutdown material, or that could otherwise cause damage to core components, 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.

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DEFINITIONS

CORE AVERAGE INLET TEMPERATURE

1.12 The CORE AVERAGE INLET TEMPERATURE shall be the arithmetic average of the operating circulator inlet temperatures, adjusted for circulator power input, steam generator regenerative heat input, and heat transfer to (or from) the PCRV liner cooling system.

CORE AVERAGE OUTLET TEMPERATURE

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

CORE AVERAGE TEMPERATURE

1.14 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, CORE AVERAGE TEMPERATURE shall be thermodynamically calculated based on CORE AVERAGE INLET and CORE AVERAGE OUTLET TEMPERATURES, PRIMARY COOLANT FLOW, and reactor power.

DOSE EQUIVALENT I-131

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

E-BAR - AVERAGE DISINTEGRATION ENERGY

1.16 E-BAR shall be the average (weighted in proportion to the concentration of each noble gas radionuclide in the sample) of beta plus gamma energy per disintegration (MeV/d) for the noble gas radionuclides in the sample. The E-BAR determination shall include quantitative measurement of the 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.

FREQUENCY NOTATION

1.17 The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals defined in Table 1.2, or as otherwise specified in the surveillance requirement.

INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE

1.18 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 its COMPARISON REGION as determined from physics calculations,

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

3. The measured refueling region outlet temperature of its COMPARISON REGION.

IRRADIATED FUEL

1.19 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)

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

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MEMBER(S) OF THE PUBLIC

1.21 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

1.22 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety 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 safety function(s) are also capable of performing their related support function(s). Nonessential portions of a system, subsystem, train, component or device need not be operational provided that the specified safety function is maintained.

OPERATIONAL MODE- MODE

1.23 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 1.1.

PHYSICS TESTS

1.24 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)

1.25 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 withdrawal prohibit functions, as addressed in Specification 3/4.3.1.

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POWER-TO-FLOW RATIO (P/F)

1.26 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

1.27 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.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant of 842 MWt.

REACTOR BUILDING CONFINEMENT INTEGRITY

1.29 REACTOR BUILDING CONFINEMENT INTEGRITY shall exist when:

a. The reactor building louvers are closed.

b. Either the truck doors to the truck bay and the personnel access door in the truck door are closed, or the truck bay floor hatch, the truck bay overhead sliding hatch, and the internal personnel door in the truck bay are closed.

REACTOR PRESSURES

1.30 REACTOR PRESSURES shall be defined as:

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 above the NORMAL WORKING PRESSURE.

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) varies in the POWER range between 610 psia and 700 psia (598 psig and 688 psig). See Figure 2.2.1-2.

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REFUELING CYCLE

1.31 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. An "n" refueling follows the REFUELING CYCLE with the same number - i.e., the sixth refueling follows the sixth REFUELING CYCLE. A REFUELING CYCLE Surveillance shall be performed at least once per 18 months.

REPORTABLE EVENT

1.32 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 or 50.73 of 10 CFR Part 50.

SAFE SHUTDOWN COOLING

1.33 SAFE SHUTDOWN COOLING shall be the removal of core stored energy and decay heat by forced circulation, using SAFE SHUTDOWN COOLING equipment. This equipment includes those systems and components involved in supplying firewater to the steam generators and helium circulator water turbine drives, as described in Specification 3/4.5. The reactivity condition in the core during SAFE SHUTDOWN COOLING is subcritical.

SAFETY LIMIT

1.34 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

1.35 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 capable of being withdrawn, which is assumed to be fully withdrawn.

SITE BOUNDARY

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

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STAGGERED TEST BASIS

1.37 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 sub-intervals, and

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

SURVEILLANCE INTERVAL

1.38 The SURVEILLANCE INTERVAL shall be the nominal period of time for the performance of surveillance requirements. Specific intervals are identified in Table 1.2.

THERMAL POWER

1.39 THERMAL POWER shall be the total reactor core heat transferred to the reactor coolant, as determined by an appropriate heat balance calculation, or from calibrated nuclear instrumentation.

THREE ROOM CONTROL COMPLEX

1.40 The THREE ROOM CONTROL COMPLEX shall be that area of the turbine building which includes the control room, the auxiliary electric equipment room, and the 480 volt switchgear room.

TRIP

1.41 TRIP shall be the switching of an instrument or a device from its normal state to a state such that it is performing its protective function. The result of a TRIP on a system level may be control rod scram, pressure relief, loop shutdown, etc.

TRIP SETPOINT

1.42 The TRIP SETPOINT shall be the least conservative "as left" value (as indicated) on a protective device to prevent a measured quantity from exceeding the ALLOWABLE VALUE.

UNRESTRICTED AREA

1.43 An UNRESTRICTED AREA shall be any area inside or outside 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.

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TABLE 1.1

OPERATIONAL MODES

MODE	INTERLOCK SEQUENCE SWITCH SETTING	REACTOR MODE SWITCH SETTING	% RATED THERMAL POWER*
POWER (P)	Power	Run	> 30%
LOW POWER (L)	Low Power @	Run	> 5% and ≤ 30%
STARTUP (S/U)	Startup	Run	≤ 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.

\# The Reactor Mode Switch setting may be changed for the purpose of performing surveillances or other tests, provided the control rods are verified to remain fully inserted (or as otherwise required for Refueling operations or surveillance testing) by a second licensed operator or other qualified member of the unit technical staff.

@ The Interlock Sequence Switch setting may be changed to the POWER position and power may be increased to no greater than 40%, for the purpose of performing surveillances or other tests, for up to 72 hours, without being considered a change in OPERATIONAL MODES.

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TABLE 1.2

FREQUENCY NOTATION

FREQUENCY NOTATION	SURVEILLANCE INTERVAL
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days
SA	At least once per 184 days
A	At least once per 366 days
R	At least once per 18 months
P	Prior to each reactor startup, if not performed within previous 7 days

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SECTION 2.0

SAFETY LIMITS and LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE SAFETY LIMIT

2.1.1 The Integral Fraction of Allowable Operating Time with POWER-TO-FLOW RATIOS that exceed the limits of Figure 3.2.6-1 shall not exceed 1.0 during the lifetime of any fuel segment.

APPLICABILITY: POWER and LOW POWER*

ACTION: With the above SAFETY LIMIT exceeded:

a. Be in at least SHUTDOWN within 24 hours, and

b. Comply with the requirements of Specification 6.7.

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* Applicable only above 15% RATED THERMAL POWER.

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BASIS FOR SPECIFICATION SL 2.1.1

SAFETY LIMIT 2.1.1 limits the P/F Integral Fraction of Allowable Operating Time of the summation of a number of individual transients. The individual transients are limited by Specification 3.2.6. The BASIS for Specification 3.2.6 is also applicable to SAFETY LIMIT 2.1.1. Further discussion on the reactor core SAFETY LIMIT is provided in FSAR section 3.6.8.

To ensure fuel particle integrity 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, with the higher temperature being at the center of the fuel rod and the lower temperature at the outer edge of the fuel rod. 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 established to ensure 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 coolant loop.

As stated in LCO 3.2.6, the Integral Fraction of Allowable Operating Times is determined as follows:

1. The range of possible POWER-TO-FLOW RATIOS above the limit of Figure 3.2.6-1 is divided into intervals, for ease of calculation.

2. The Allowable Operating Time above the limit of Figure 3.2.6-1 is determined each P/F RATIO interval from Figure 3.2.6-2.

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BASIS FOR SPECIFICATION SL 2.1.1 (Continued)

3. For each transient, the actual time period during which the P/F RATIO exceeded the limit of Figure 3.2.6-1 for each interval is divided by the Allowable Operating Time for that interval.

4. The individual fractions determined in Step 3 above are summed for each fuel segment, over its lifetime in the core. This is the Integral Fraction of Allowable Operating Time which may not exceed 1.0.

APPLICABILITY is limited to power levels above 15% RATED THERMAL POWER, in that Figure 3.2.6-1 covers only the range of 15% to 100% power. Specification 3.2.4, Core Inlet Orifice Valves/Minimum Helium Flow and Maximum Core Region Temperature Rise, includes power levels below 15% where core temperatures are lower, and also overlaps the power levels addressed by this SAFETY LIMIT.

BASIS for Orderly Shutdown

Following determination (Specification 3.2.6 ACTION c.1) that SAFETY LIMIT 2.1.1 has been exceeded, shutdown is allowed to be performed in an orderly manner (24 hours to be in at least SHUTDOWN), thus minimizing unnecessary transient effects on other plant components. Any severe transient that significantly exceeds the limits of Specification 3.2.6 would require a much faster plant shutdown (Specification 3.2.6 ACTION b), if it did not result in a scram by automatic response of the PLANT PROTECTIVE SYSTEM

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR VESSEL PRESSURE

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

APPLICABILITY: At all times

ACTION: POWER, LOW POWER, and STARTUP

With the PCRV internal pressure or the penetration interspace pressure exceeding 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. .

SHUTDOWN and REFUELING

With the PCRV internal pressure or penetration interspace pressure exceeding 845 psig, reduce the pressure to within its limit within 1 hour, and comply with the requirements of Specification 6.7.

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BASIS FOR SPECIFICATION SL 2.1.2

The restriction of this SAFETY LIMIT protects the PCRV and its penetrations from overpressurization, and thereby ensures the integrity of the PCRV as a fission product barrier.

A detailed discussion of the PCRV and penetrations, including design bases, is contained in FSAR Section 5. All pressure containing elements for the primary coolant system are designed for a pressure equal to the PCRV Reference Pressure of 845 psig. (FSAR Section 5.8.2). From initial reactor startup, after completion of the initial proof test, pressurization of the PCRV above Reference Pressure is 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 consistent with the design criteria.

Prior to initial operation, the Fort St. Vrain PCRV was subjected to an initial proof test pressure (approximately equal to 1.15 times Reference Pressure) to demonstrate integrity, verify the structural response of the vessel to an internal pressure greater than Reference Pressure, and 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: Specification 3.9.1 contains pressure limits applicable during the handling of IRRADIATED FUEL in the reactor vessel.

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS TRIP SETPOINTS:

2.2.1 The PLANT PROTECTIVE SYSTEM (PPS) 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 3.6.1.1

ACTION: a. With a LIMITING SAFETY SYSTEM SETTING less conservative than the value shown in the ALLOWABLE VALUE column of Table 2.2.1-1, except as permitted by ACTION b below, declare the channel inoperable and apply the applicable ACTION requirement of Specification 3.3.1 (PPS) or 3.6.1.1 (PCRV Pressurization).

b. With a Linear Channel-High Neutron Flux channel setpoint less conservative than the value shown in the ALLOWABLE VALUE column of Table 2.2.1-1 due to a change in power level that requires a change in TRIP SETPOINT, adjust the TRIP SETPOINT as required within 12 hours or be in at least SHUTDOWN within the following 24 hours.

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Table 2.2.1-1

LIMITING SAFETY SYSTEM SETTINGS

PARAMETER	FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Core			
a) Linear Channel-High (Neutron Flux)	Scram	Varies as a Function of Indicated THERMAL POWER per Figure 2.2.1-1	Varies as a Function of Indicated THERMAL POWER per Figure 2.2.1-1
b) Reheat Steam Temperature-High	Scram	≤ 1055 degrees F	≤ 1067 degrees F
c) Primary Coolant Pressure-Programmed Low	Scram	≤ 68.6 psi below Normal, programmed with Circulator Inlet Temperature. Upper TRIP SETPOINT of ≥ 631.1 psia, for Circulator Inlet Temperature ≥ 742 degrees F	≤ 72.7 psi below Normal, programmed with Circulator Inlet Temperature per Figure 2.2.1-2. Upper Limit to produce TRIP at ≥ 627 psia, for Circulator Inlet Temperature ≥ 742 degrees F

Table 2.2.1-1 (Continued)

LIMITING SAFETY SYSTEM SETTINGS

PARAMETER	FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. Reactor Vessel Pressure			
a) Primary Coolant Pressure-Programmed High	Scram and Preselected Loop Shutdown and Steam/Water Dump	≤ 46 psi above Normal, programmed with Circulator Inlet Temperature. Upper TRIP SETPOINT of ≤ 746.3 psia, for Circulator Inlet Temperature ≥ 742 degrees F. Lower TRIP SETPOINT of ≤ 538.3 psia for Circulator Inlet Temperature ≤ 415 degrees F.	≤ 52.7 psi above Normal, programmed with Circulator Inlet Temperature per Figure 2.2.1-2. Upper limit to produce TRIP at ≤ 753 psia, for Circulator Inlet Temperature ≥ 742 degrees F. Lower limit to produce TRIP at ≤ 545 psia for Circulator Inlet Temperature ≤ 415 degrees F.
b) Primary Coolant Moisture-High	Scram, Loop Shutdown, and Steam/Water Dump	≤ 60.5 degrees F dewpoint temperature	≤ 62.2 degrees F dewpoint temperature
c) PCRV Pressure:	Pressure Relief		
Rupture Disc (Low Set Safety Valve)		812 psig plus or minus 8 psi	820 psi

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

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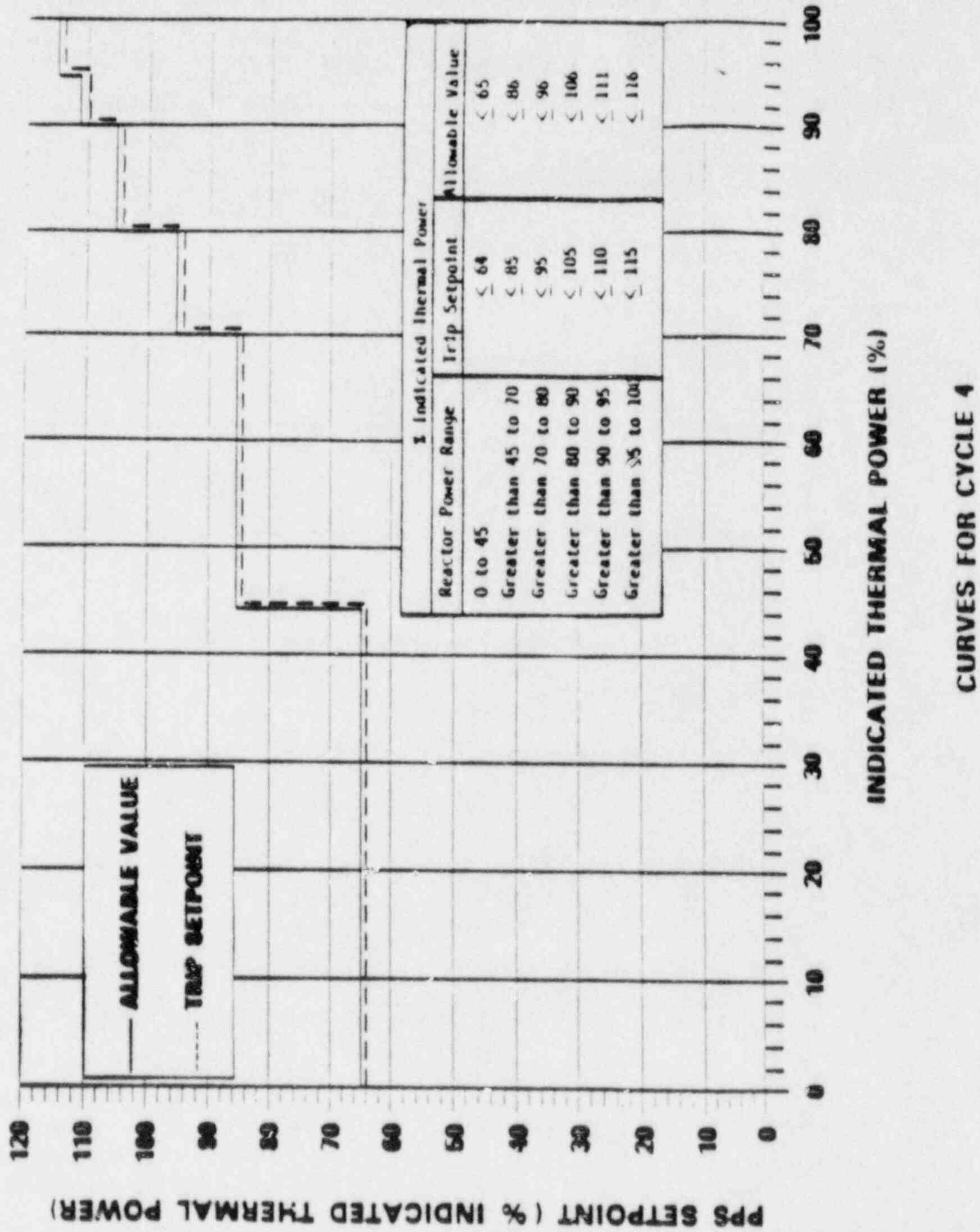

HIGH NEUTRON FLUX SCRAM
DETECTOR DECALIBRATION

Figure 2.2.1-1

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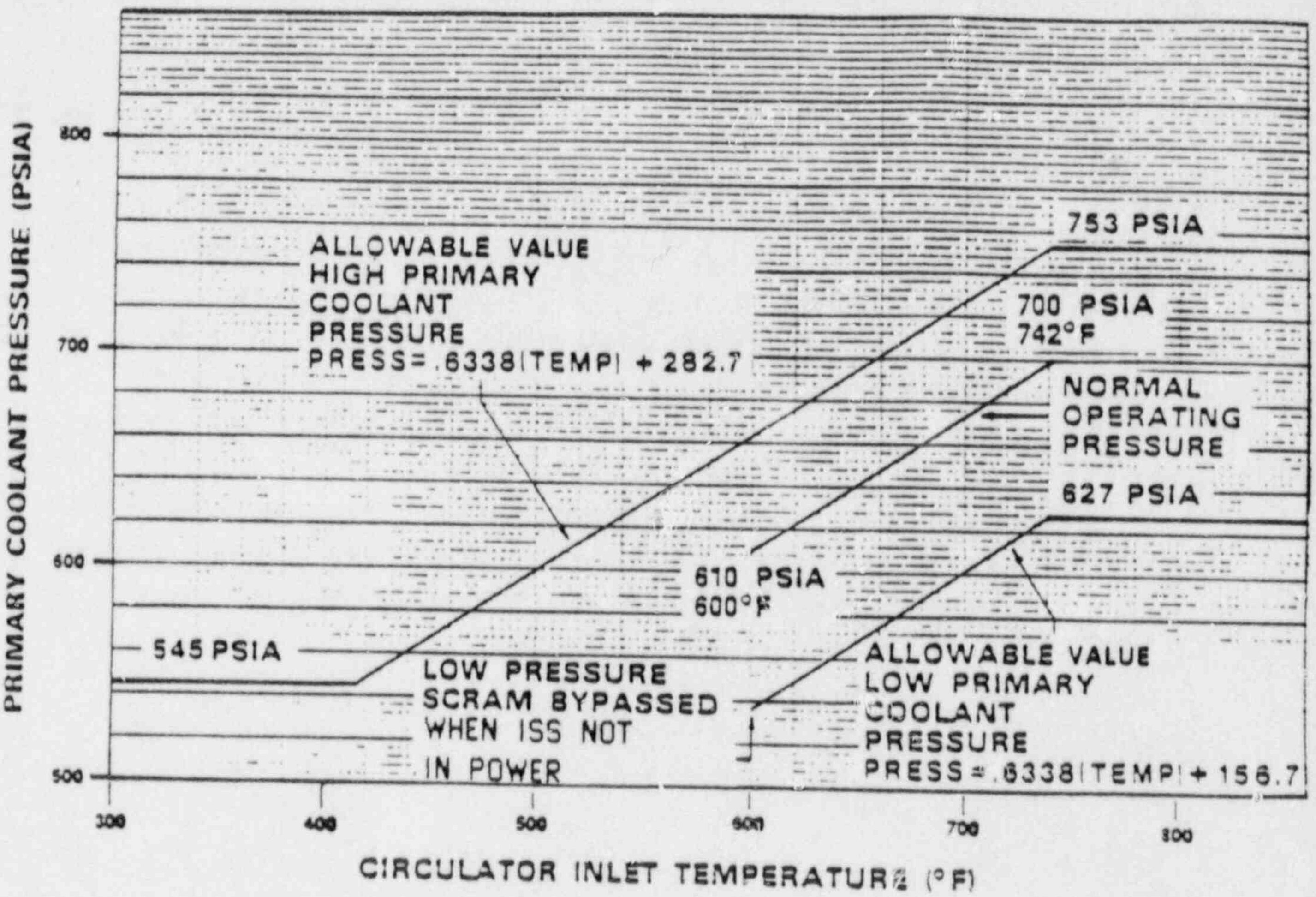

PRIMARY COOLANT PRESSURE vs. CIRCULATOR INLET TEMPERATURE
ALLOWABLE OPERATION

Figure 2.2.1-2

BASIS FOR SPECIFICATION SL 2.2.1

TRIP SETPOINTS

SAFETY LIMITS have been specified in Specification 2.1.1 and 2.1.2 to safeguard the fuel particle integrity and the reactor primary 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. This specification establishes the TRIP SETPOINTS and ALLOWABLE VALUES for these automatic protective devices.

Operation with setpoints less conservative than the TRIP SETPOINT but within the ALLOWABLE VALUE is acceptable since an allowance has been made in the safety analysis to accommodate this error, as described below. The ALLOWABLE VALUE is the threshold for REPORTABLE EVENTS. That is, if the "as found" setpoint exceeds the ALLOWABLE VALUE, it shall be reset and the requirements for REPORTABLE EVENTS are applicable.

General Methodology

The Analysis Value is the value of a parameter for which a TRIP and initiation of automatic protective action is assumed to occur in FSV accident analyses (FSAR Chapter 14). Provided that the TRIP occurs at a value equal to or more conservative than the Analysis Value, analyses demonstrate that consequences of the accident or transient are acceptable.

ISA Standard, S67.04-1982 has been applied to these Analysis Values to arrive at ALLOWABLE VALUES and TRIP SETPOINTS for each PPS parameter.

BASIS FOR SPECIFICATION SL 2.2.1 (Continued)

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 indicated as percent of RATED THERMAL POWER. For slow maneuvers, those where core THERMAL POWER, surface heat flux, and the heat 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 heat 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 heat transferred to the helium will be less than the percent increase in neutron flux. TRIP SETPOINTS that ensure a reactor scram at no greater than 140% 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 external-core neutron flux detectors. To account for this potential decalibration and other instrumentation errors, the actual TRIP SETPOINT is administratively set less than 140% RATED THERMAL POWER based upon indicated power. These administratively set flux TRIP SETPOINTS ensure the scram will occur at or less than 140% RATED THERMAL POWER for those postulated reactivity accidents evaluated in FSAR Section 14.2 Additional discussion on detector decalibration is given in FSAR Section 7.3.1.2.1.

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 an excessive POWER-TO-FLOW RATIO due to a power increase or steam flow imbalance. (FSAR Section 14.2)

BASIS FOR SPECIFICATION SL 2.2.1 (Continued)

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 a 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 increase of approximately 1 psi per second. The normal PPS 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 less than or equal to 46 psi above the Normal Operating Pressure between 25% and 100% 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 was 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 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 2.2.1-2. The PCRV safety valves provide the ultimate protection against primary coolant system pressure exceeding the PCRV Reference Pressure of 845 psig.

Primary Coolant Moisture - High

The high moisture TRIP SETPOINT corresponding to 60.5 degrees F dewpoint was established, considering the moisture monitor characteristics and the necessity to minimize water inleakage to the primary coolant 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 Primary Coolant Pressure-Programmed High. 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, 7.1.2.4, and 7.1.2.5.)

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BASIS FOR SPECIFICATION SL 2.2.1 (Continued)

PCRV Pressure

The PCRV safety valves provide the ultimate protection against primary coolant system pressure exceeding the PCRV Reference Pressure of 845 psig. This engineered safeguard system consists of the isolation valves, the rupture discs, the relief valves, and the containment tank. Two safety valves are provided, either of which is adequate to prevent exceeding the PCRV Reference Pressure in the event of a steam generator subheader rupture, which is the only credible means of substantially increasing the primary coolant pressure.

If the pressure in the PCRV were to rise significantly above the Normal Working Pressure, the low-set rupture disc would rupture within the range of 804 to 820 psig (812 psig +/- 1%). The low set safety valve, set at 796 psig plus or minus 1%, would be wide open and relieving at 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 relieving at 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, due to line losses.

The minimum permissible TRIP SETPOINT of each PCRV overpressure relief train rupture disc and relief valve is specified to provide assurance that primary coolant helium will not be vented to atmosphere during primary coolant pressure surges, resulting from transients or accidents, in which pressures do not approach the ALLOWABLE VALUE and thereby do not challenge the integrity of the PCRV. (FSAR Section 6.8.3.) See Specification 3.6.1.1.

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BASIS FOR SPECIFICATION SL 2.2.1 (Continued)

Helium Circulator Penetration Interspace Pressure

The penetration interspaces are protected against pressures exceeding PCRV Reference Pressure (845 psig). 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 808 to 842 psig (825 psig +/- 2%). The safety valves would open in the range of 781 to 829 psig (805 psig +/- 3%). The safety valves would reseat at about 765 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 overpressure protection.

The minimum permissible TRIP SETPOINT of each rupture disc and associated relief valve is specified to provide assurance that PCRV penetration interspace helium, which could potentially be radioactive, will not be vented to atmosphere during interspace pressure surges in which pressures do not approach the ALLOWABLE VALUE and thereby do not challenge the integrity of the PCRV penetration. (FSAR Section 5.8.2). See Specification 3.6.1.2.

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 Pressure (845 psig). A redundant safety valve and rupture disc are provided. The rupture discs would burst in the pressure range of 808 to 842 psig (825 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.

The minimum permissible TRIP SETPOINT of each rupture disc and associated relief valve is specified to provide assurance that PCRV penetration interspace helium, which could potentially be radioactive, will not be vented to atmosphere during interspace pressure surges in which pressures do not approach the ALLOWABLE VALUE and thereby do not challenge the integrity of the PCRV penetration. (FSAR Section 5.8.2). See Specification 3.6.1.2.

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SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

3.0 LIMITING CONDITIONS FOR OPERATION

3.0.1 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.

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

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, ACTION shall be initiated within 1 hour to place the unit in an OPERATIONAL MODE in which the specification does not apply by placing it, as applicable, in at least LOW POWER within the next 12 hours and in at least SHUTDOWN within the following 12 hours. This requirement is not applicable in SHUTDOWN or REFUELING.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements, or as required by automatic or manual protective ACTION. Exceptions to these requirements are stated in the individual specifications.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.0.5 These specifications identify equipment redundancy and forced circulation requirements in terms of whether the CALCULATED BULK CORE TEMPERATURE (CBCT) is greater or less than 760 degrees F.

Where referenced, the time at which the CBCT reaches 760 degrees F following an interruption (assumed or actual) of all PRIMARY COOLANT FLOW determines when specification requirements apply. If the specification is applicable when the CBCT exceeds 760 degrees F, the requirements apply after this time. If the specification is applicable when the CBCT is less than 760 degrees F, the requirements apply before this time.

The time for the CALCULATED BULK CORE TEMPERATURE to reach 760 degrees F following an interruption of all PRIMARY COOLANT FLOW is determined as follows:

a. Using the applicable operating power history prior to interruption of PRIMARY COOLANT FLOW, determine the decay heat power (MW) from Figure 3.0-1.

b. Using the average core temperature prior to the PRIMARY COOLANT FLOW interruption, determine the decay heat energy (MW-Hr) required to raise the CBCT to 760 degrees F, from Figure 3.0-2.

c. Using the decay heat power (MW) from Figure 3.0-1, and the decay heat energy (MW-Hr) from Figure 3.0-2, determine the time required to reach 760 degrees F (Hr). Refer to the example on Figure 3.0-2.

d. The maximum time for which PRIMARY COOLANT FLOW can be interrupted is the time interval determined in Specification 3.0.5.c, not to exceed 21 days.

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 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.

4.0.2 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.

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

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated SURVEILLANCE INTERVAL. 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.

4.0.5 Surveillance Requirements for inservice inspection and testing of essential components shall be applicable as follows:

a. Inservice inspection and testing of essential systems and components shall be performed in accordance with the Fort St. Vrain Inservice Inspection and Testing Program, as discussed in Administrative Control 6.18.

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b. SURVEILLANCE INTERVALS specified in the Fort St. Vrain Inservice Inspection and Testing Program shall be applicable as follows in these Technical Specifications:

Fort St. Vrain Inservice Inspection and Testing Program terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

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

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

e. Nothing in the ASME Boiler and Pressure Vessel Code or the Fort St. Vrain Inservice Inspection and Testing Program shall be construed to supercede the requirements of any Technical Specification.

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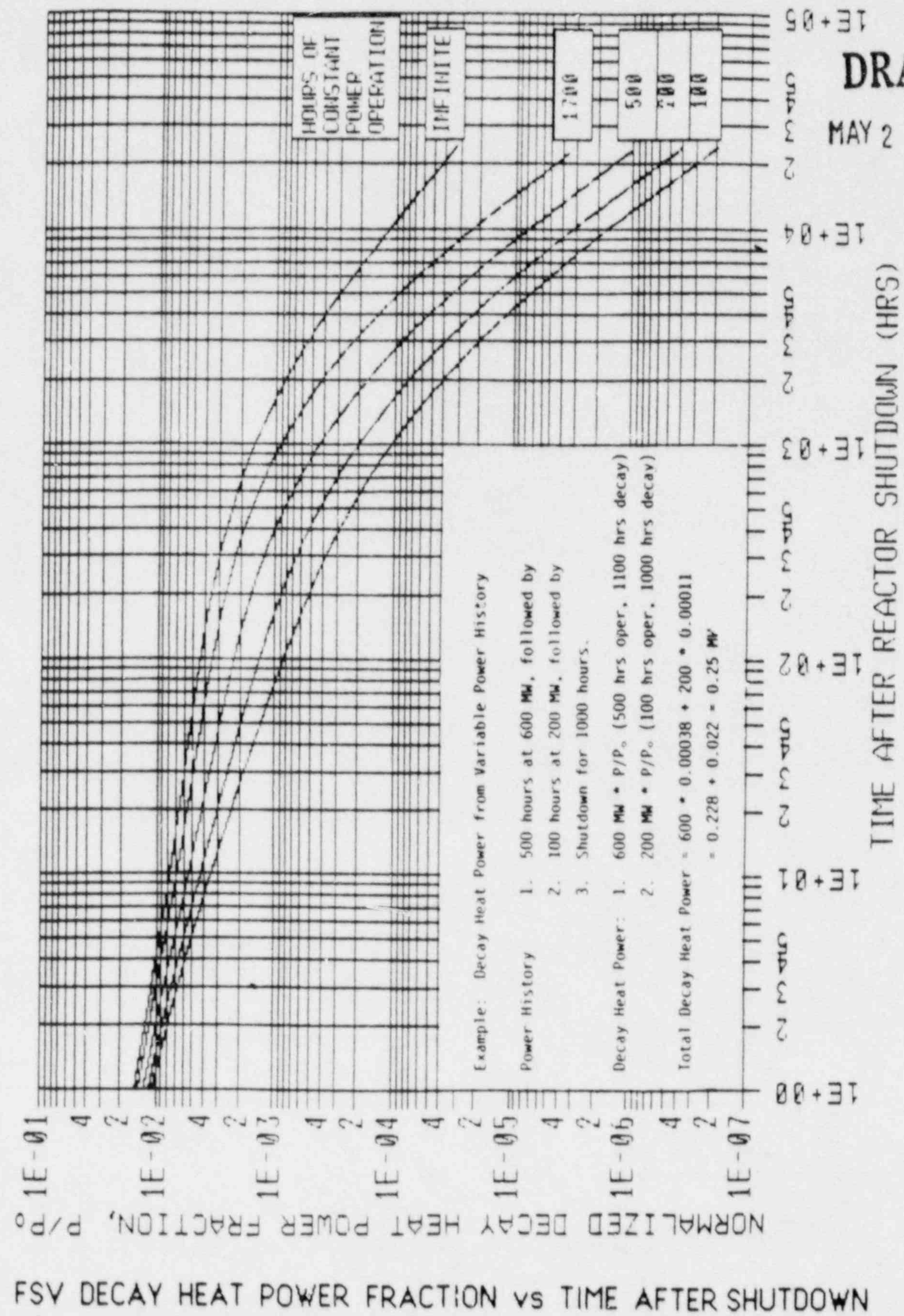
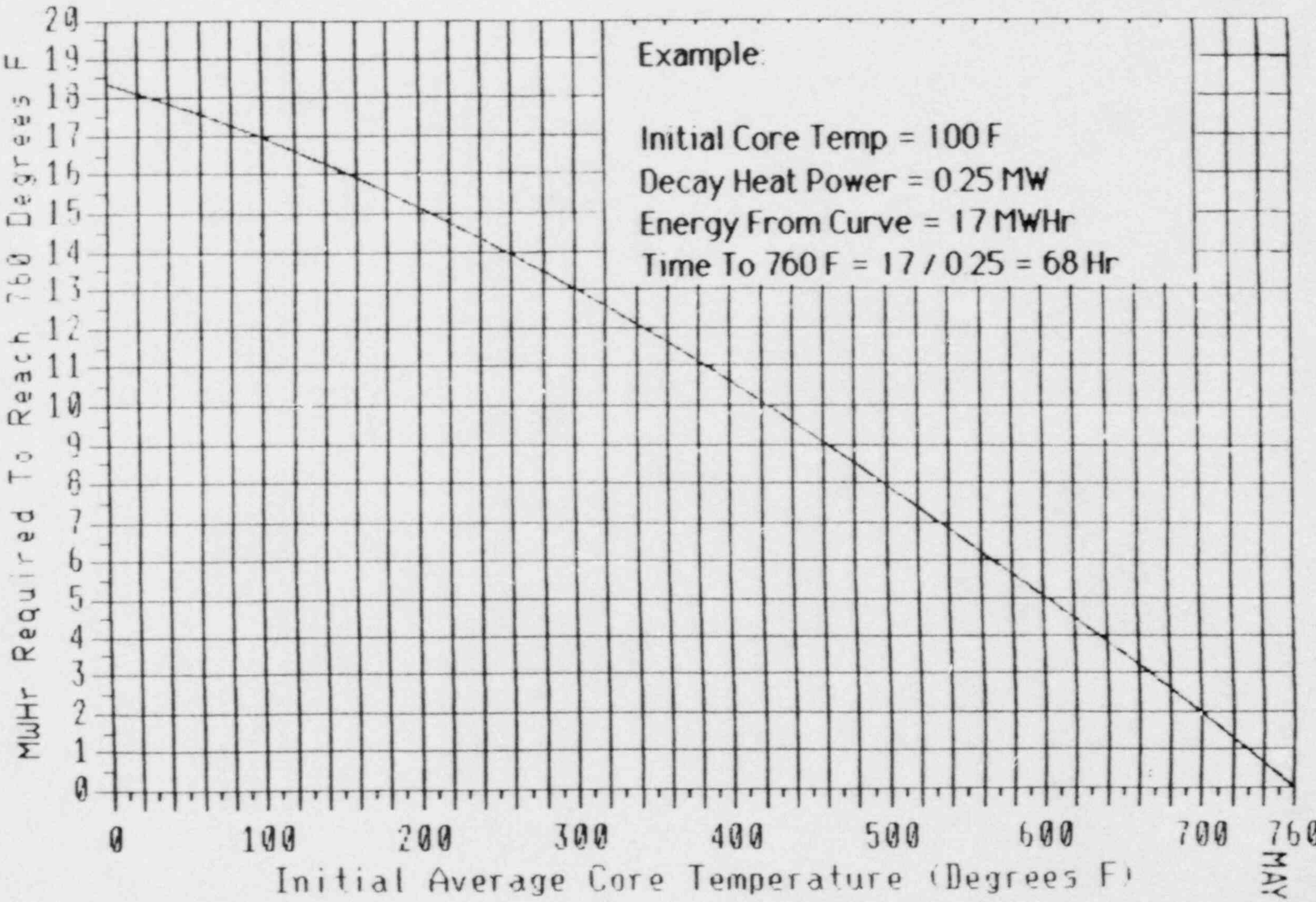


FSV DECAY HEAT POWER FRACTION vs TIME AFTER SHUTDOWN
FOR VARIOUS TIMES AT CONSTANT POWER

Figure 3.0-1



DECAY HEAT ENERGY REQUIRED TO RAISE
CORE TEMPERATURE TO 760 DEGREES F

Figure 3.0-2

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

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

These Limiting Conditions for Operation provide for operation with sufficient redundancy and/or diversity to meet the single-failure criterion as relied upon in the plant's safety analysis. The 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.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. An example of this is the ACTION to be taken for inoperable seismic monitors. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

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

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

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

This concept also applies to progressive ACTIONS. For example, if an ACTION allows 72 hours to repair one SLRDIS valve and 24 hours to repair all but one SLRDIS valve (in the event several valves are inoperable), once the equipment is restored to only one inoperable valve the original 72 hour clock is applicable.

3.0.3 This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.7.3 requires two instrument air systems to be OPERABLE and provides explicit ACTION requirements if only one instrument air system is OPERABLE. Under the requirements of Specification 3.0.3, if both of the instrument air systems are inoperable, measures must be initiated within 1 hour to place the unit in at least LOW POWER within the next 12 hours and in at least SHUTDOWN within the following 12 hours.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

The purpose of this specification is to delineate the time limits for placing the unit in a SHUTDOWN MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

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

The time limits of Specification 3.0.3 allow 24 hours for the plant to be in the SHUTDOWN MODE when a shutdown is required during plant operation. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach SHUTDOWN, or other applicable MODE, is not reduced. For example, if LOW POWER is reached in 8 hours, the time allowed to reach SHUTDOWN is the next 16 hours because the total time to reach SHUTDOWN is not reduced from the allowable limit of 24 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred for having reached a lower MODE of operation in less than the total time allowed.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

The ACTION to be in LOW POWER in 12 hours and in SHUTDOWN in the following 12 hours defines an orderly shutdown at Fort St. Vrain. 12 hours to reduce to LOW POWER (30%) is allowed to minimize unnecessary transients on the steam generator tubing that would result from going through boilout (approximately 18-22%) during reductions to lower power levels. The process of reducing power in an orderly manner from less than 30% (LOW POWER) to SHUTDOWN is complicated and time consuming in that all of the core orifices must be adjusted from an equal temperature configuration to an equal flow configuration, which requires approximately 4 to 6 hours. Orifice adjustments are continuously performed during a power reduction and the change in configuration is initiated about 12-14% power. In addition, the auxiliary boiler(s) is brought on-line to provide sufficient drive capability for the helium circulators when adequate nuclear generated steam is not available (approximately 8% power).

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

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability conditions must be made with: (1) the full compliment of required systems, equipment, or components OPERABLE and (2) all other parameters as specified in the Limiting Condition for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.0.5 The CALCULATED BULK CORE TEMPERATURE (CBCT) is used in the FSV Technical Specification as an indicator of decay heat levels that determines the applicability of LCO or ACTION requirements. A CBCT below 760 degrees F indicates relatively low decay heat and stable core conditions. This generally is the case in REFUELING, in SHUTDOWN (except for a period just after operation), and for a period in STARTUP after an extended shutdown condition. As long as the CBCT remains below 760 degrees F, forced circulation is not required and redundancy in SAFE SHUTDOWN COOLING equipment is also not required.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

The CALCULATED BULK CORE TEMPERATURE is the calculated, time dependent, average temperature of the core, including graphite and fuel, but not the reflector, assuming an interruption of all forced circulation of PRIMARY COOLANT FLOW. The calculation uses several conservative assumptions: 1) The decay heat power at the start of the core heatup has been conservatively selected using Figure 3.0-1 and is assumed to remain constant for the total interval; 2) All decay heat power generated is assumed to be retained in the active core with no heat transfer to the reflector, PCRV internals or primary coolant; and 3) A 10 percent margin has been included on the core heatup time given in Figure 3.0-2. If the active core remains below 760 degrees F, which corresponds to the design maximum core inlet temperature, then there can be no damage to fuel or PCRV internal components, even in the absence of forced circulation of primary coolant helium flow.

The time required to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F is primarily dependent upon the decay heat power and the current average core temperature. This time is conservatively estimated using the data in Figures 3.0-1 and 3.0-2. The decay heat power data in Figure 3.0-1 was explicitly calculated for the Fort St. Vrain core and is derived from Appendix D.1 of the FSAR, Figure D.1-9, revision 2. The decay heat power resulting from a varying power history can be conservatively calculated by representing the actual power history by a series of constant power steps and then summing the individual decay heat power contribution from each power step. The decay heat power due to operation during the last 1000 days can be determined in this manner. Residual decay heat power from earlier operation is conservatively estimated by assuming that this was full power continuous operation, and by then adding this decay heat power component to the calculated decay heat power value.

Knowing the decay heat power and the current average core temperature, the time for the core to heat up from its current temperature to 760 degrees F can be obtained from Figure 3.0-2, which has been generated using the adiabatic heat transfer model and a heat capacity for composite graphite as given in Appendix D.1 of the FSAR, Figure D.1-3, revision 2.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

To allow for uncertainties associated with determining the time to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, an additional 10 percent has been included in the decay heat energy given in Figure 3.0-2. In addition, it has been specified that any time interval for which the PRIMARY COOLANT FLOW is interrupted shall not exceed 21 days. This ensures a restoration of forced circulation of PRIMARY COOLANT FLOW to confirm core average temperature on a periodic basis. Although much longer intervals can be determined from Figure 3.0-2, 21 days is an adequate time to conduct operations requiring flow interruption, such as maintenance or circulator changeout. Operating experience at Fort St. Vrain has shown that the calculated core heatup rate has always been higher than the actual core heatup rate.

4.0.1 The Surveillance Requirements specified in these Technical Specifications define the tests, calibrations, and inspections which ensure the performance and OPERABILITY of equipment essential to safety or equipment required to prevent or mitigate the consequences of abnormal situations.

These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

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

This specification provides that surveillance activities necessary to ensure that the Limiting Conditions for Operation are being met and that they will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable.

Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal SURVEILLANCE INTERVAL. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under these criteria, equipment, systems, or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed SURVEILLANCE INTERVAL and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed SURVEILLANCE INTERVAL was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed SURVEILLANCE INTERVAL, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to possible enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a REPORTABLE EVENT under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed, unless a longer exception is specifically allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

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

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASIS FOR SPECIFICATION LCO 3.0/SR 4.0 (Continued)

Under the terms of this specification, for example, during initial plant STARTUP or following extended plant outages, the applicable surveillance activities must be performed within the stated SURVEILLANCE INTERVAL prior to placing or returning the system or equipment into OPERABLE status.

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

4.0.5 This specification ensures that inservice inspection and testing of essential systems and components will be performed in accordance with the periodically updated Fort St. Vrain Inservice Inspection and Testing Program, as discussed in Administrative Control 6.18.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by the Fort St. Vrain Inservice Inspection and Testing Program. This clarification is provided to ensure consistency in SURVEILLANCE INTERVALS throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the Fort St. Vrain Inservice Inspection and Testing Program. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the Inservice Inspection and Testing Program provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the Inservice Inspection and Testing Program provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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REACTIVITY CONTROL SYSTEMS

3/4.1.1 CONTROL ROD PAIR OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.1 All control rod pairs not fully inserted shall be OPERABLE with:

a. A scram time less than or equal to 152 seconds from the fully withdrawn position,

b. A control rod drive (CRD) motor temperature less than or equal to 250 degrees F,

c. A helium purge flow not carrying condensed water to each CRD penetration when reactor pressure is above 100 psia, and

d. The absence of a slack cable alarm.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION:

a. With one or more control rod pairs inoperable due to being immovable (i.e., not capable of being fully inserted), within 10 minutes initiate a reactor shutdown and an assessment of the SHUTDOWN MARGIN, and be in at least SHUTDOWN within the next 12 hours.

b. With one control rod pair inoperable due to having a scram time greater than 152 seconds, operation may continue provided that within 24 hours:

1. The control rod pair is restored to OPERABLE status, or

2. The control rod pair is fully inserted, or

3. The SHUTDOWN MARGIN requirement of Specification 3.1.3 is satisfied with the control rod pair considered inoperable in its present position.

If none of the above conditions can be met, be in at least SHUTDOWN within the next 12 hours.

SPECIFICATION LCO 3.1.1 (Continued)

c. With two or more control rod pairs inoperable due to having a scram time greater than 152 seconds, immediately initiate a reactor shutdown and be in at least SHUTDOWN within 12 hours.

d. With one or more control rod pairs having a CRD motor temperature greater than 250 degrees F, operation may continue provided that within 24 hours:

1. The control rod pair(s) is restored to OPERABLE status, or

2. Surveillance testing per Specification 4.1.1. b is performed on the control rod pair(s) once every 24 hours when the CRD motor temperature exceeds 250 degrees F. With one or more control rod pairs exceeding a scram time of 152 seconds, comply with ACTIONS b or c above. With scram times less than or equal to 152 seconds, up to four control rod pairs with CRD motor temperatures greater then 250 degrees F may be considered OPERABLE for SHUTDOWN MARGIN determination.

e. With no purge flow to one CRD penetration, operation may continue provided that within 24 hours:

1. Purge flow is restored to the CRD penetration, or

2. The control rod pair is fully inserted, or

3. The SHUTDOWN MARGIN requirement of Specification 3.1.3 is satisfied with the control rod pair considered inoperable in its present position.

If one of the above conditions cannot be met, be in at least SHUTDOWN within 12 hours.

f. With no purge flow to two or more CRD penetrations:

1. Restore purge flow within 2 hours, or

2. Be in at least SHUTDOWN within the next 12 hours.

g. With the water level in the knock-out pot for the CRD purge flow lines greater than 6 inches, but with the knock-out pot not flooded:

1. Within 1 hour drain the knock-out pot and establish a helium purge flow not carrying condensed water, or

2. Be in at least SHUTDOWN within the next 12 hours.

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SPECIFICATION LCO 3.1.1 (Continued)

h. With the knock-out pot for the CRD purge flow lines flooded:

   1. Be in at least SHUTDOWN within the next 12 hours, and

   2. Perform surveillance SR 4.1.6.2.d.4.

i. With a slack cable alarm, within 24 hours determine whether a slack cable condition exists (i.e., a parted cable, detached cable, or failed instrumentation that is inaccessible for repair during operation). If an actual slack cable condition exists, be in at least SHUTDOWN within the next 24 hours. If the alarm is due to some other condition, restore the alarm to OPERABLE status within the next 24 hours.

j. The provisions of Specification 3.0.4 are not applicable for changes between STARTUP, LOW POWER, and POWER. Prior to entry into STARTUP from SHUTDOWN, all requirements of this LCO must be met, without reliance on provisions contained in the ACTION statements.

SURVEILLANCE REQUIREMENTS

4.1.1 Each control rod pair shall be demonstrated OPERABLE:

a. Prior to withdrawal of control rod pairs to achieve criticality (if not performed in the previous 7 days) by performing a partial scram test of at least 10 inches on all control rod pairs being withdrawn, and verifying that the extrapolated scram time is less than or equal to 152 seconds.

b. At least once per 24 hours by:

   1. Verifying that all CRD motor temperatures are less than or equal to 250 degrees F.

      a) With one or more CRD motor temperature(s) exceeding 215 degrees F:

         1) The temperature of any CRD motor exceeding 215 degrees F shall be recorded,

SPECIFICATION SR 4.1.1 (Continued)

2) A partial scram test as described in Specification 4.1.1.c shall be performed at least once per 24 hours on the control rod pair with the highest motor temperature and for all control rod pairs greater than 250 degrees F, and

3) A report on the partial scram test results and the maximum daily temperature of any control rod pairs with motor temperatures exceeding 215 degrees F shall be submitted to the NRC once every 31 days.

b) If CRD motor temperature instrumentation is inoperable, an engineering evaluation shall be performed to determine CRD motor temperature by comparison.

2. Verifying purge flow to each CRD by verifying flow in each subheader, when reactor pressure is above 100 psia; and

3. Verifying that the purge flow is not carrying condensed water by verifying that the water level in the knock-out pot is less than 6 inches.

4. Verifying that the slack cable alarm is not actuated.

c. At least once per 7 days by:

1. Performing a partial scram test of at least 10 inches on all partially inserted and fully withdrawn control rod pairs, except the regulating rod pair, and verifying that the extrapolated scram time is less than or equal to 152 seconds; and

2. Performing a partial scram test of approximately 2 inches on the regulating rod pair and verifying control rod pair movement.

d. Prior to withdrawal of any control rod pair (if not performed in the previous 7 days) by performing a partial scram test of at least 10 inches and verifying that the control rod pair inserts freely and can be considered scrammable.

e. During each shutdown of 10 days or longer (if not performed during the previous 31 days) by performing a full stroke scram test on all control rod pairs and verifying a scram time less than or equal to 152 seconds.

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SPECIFICATION SR 4.1.1 (Continued)

f. Following any maintenance on a CRD mechanism which could affect the control rod pair scram time, by performing a full stroke scram test and verifying a scram time of less than or equal to 152 seconds.

g. At least once per REFUELING:

1. By performing a CHANNEL CALIBRATION and a CHANNEL FUNCTIONAL TEST of the eight subheader CRD purge flow measurement channels,

2. By performing a CHANNEL FUNCTIONAL TEST of the CRD motor temperature and cavity temperature instrumentation,

3. By performing preventive maintenance on each CRD in a scheduled sequence such that none of the drives installed in the reactor will have gone more than 6 REFUELING CYCLES without receiving preventive maintenance. During these 6 REFUELING CYCLES, no CRD shall be in regulating rod pair service (without receiving preventive maintenance) for more than one REFUELING CYCLE. The preventive maintenance shall consist of inspecting and replacing as necessary the CRD gears, bearings, brake pads, cables, and position instrumentation, and

4. By performing a CHANNEL CALIBRATION of the CRD motor and cavity temperature instrumentation on those CRDs undergoing preventive maintenance as described in Specification 4.1.1.g.3 above.

BASIS FOR SPECIFICATION LCO 3.1.1/SR 4.1.1

Control rod pair OPERABILITY ensures that a minimum SHUTDOWN MARGIN is capable of being maintained.

The control rod pair withdrawal accident analyses described in FSAR Sections 14.2.2.6 and 14.2.2.7 were performed assuming a scram insertion time of 152 seconds and a ramp reactivity insertion of 0.080 delta k and 0.058 delta K, respectively.

Requiring the scram time to be less than or equal to 152 seconds will ensure that the ramp reactivity rate is consistent with that assumed in the accident analyses. The full insertion scram time can be determined either directly from a full insertion scram test or indirectly from a partial scram test of 10 inches or more. For the partial scram test, the estimate of an extrapolated scram time of less than or equal to 152 seconds is always based on assuming a scram from the fully withdrawn position and not from the actual rod position.

The total calculated reactivity worth of all 37 control rod pairs is 0.210 delta k, which is significantly greater than the scram reactivities assumed in the accident analyses. Therefore, a single control rod pair with a scram time greater than 152 seconds, as allowed in ACTION b of the specifications, will have no impact on the calculated consequences of the control rod pair withdrawal accident.

Temperature Limitation

Control Rod Drive Mechanism (CRDM) qualification tests were performed in a 180 degree F helium environment. The motor and brake were energized and deenergized in severe duty cycles up to once every 5 seconds for 630,000 jog cycles and 5000 scrams of the CRDM. CRDM motor temperatures ranged from 200 degrees F to 230 degrees F with an average of 215 degrees F during these tests. During power ascension testing, CRDM temperatures up to 213 degrees F were experienced at power levels up to 70%. Using data obtained during power ascension testing, a CRDM temperature of 260 degrees F was predicted for 100% power conditions with an orifice valve fully closed. The minimum predicted open position for an orifice valve at 100% power is about 10%, for which the predicted CRDM temperature is 250 degrees F. Tests conducted to 100% power indicated these predictions to be conservative because the maximum measured CRDM motor temperature was 218 degrees F. The operating temperature of the CRDM is limited by the motor insulation which is derated to 272 degrees F to account for motor temperature rise, frictional torque increase, and winding life expectancy. See Section 3.8.1.1 of the FSAR.

BASIS FOR SPECIFICATION LCO 3.1.1/SR 5.1.1 (Continued)

CRDM motor temperatures are monitored to verify that they are less than or equal to 250 degrees F. CRDM motor temperatures are alarmed at 215 and 250 degrees F, and are recorded on a multi-point recorder when they exceed 215 degrees F (FSAR 3.8.1.1). This recorder provides frequent monitoring (at least one reading per minute) and the data is retrievable as required. Any CRDM with a motor temperature greater than 215 degrees F shall be recorded every 24 hours to document that the temperature is less than 250 degrees F. In addition, the partial scram test frequency is increased from once per 7 days to once per 24 hours on the control rod pair with the highest motor temperature. A partial scram test will be performed once every 24 hours on all control rod pairs with motor temperatures exceeding 250 degrees F, to verify that the extrapolated scram time is less than or equal to 152 seconds. Verifying a control rod pair extrapolated scram time of less than or equal to 152 seconds, will ensure CRDM reliability with a motor temperature greater than 250 degrees F. These surveillances ensure that CRDM motor temperatures exceeding 250 degrees F do not degrade the CRDM's reliability to perform its design function when required, and up to four of these control rod pairs may be considered OPERABLE in SHUTDOWN MARGIN determination.

If the CRDM motor temperature instrumentation is inoperable, an engineering evaluation will be completed within 24 hours from the time the instrumentation is found to be inoperable to verify that the CRDM motor temperature is currently less than 250 degrees F. Additional temperature instrumentation located on the underside primary closure plate and the orifice valve motor plate can be used to infer the CRDM motor temperature by comparing these temperatures with those on another CRDM in a similar region. Other factors such as orifice valve position and historical temperature data may be used to determine CRDM motor temperatures by comparison.

Purge Flow

The purge flow into the CRD assembly limits the upward flow of contaminated primary system helium coolant. Purge flow to each CRD penetration is ensured by verifying that purge flow is maintained to each subheader and by sealing all the valves between the subheaders and the CRD penetrations in an open position.

A knock-out pot, moisture element, and pressure transmitter are installed in the CRD purge line, between the purified helium header and the CRD purge flow valve (FSAR 3.8.1.1). Just before the knock-out pot, an independent source of dry helium is connectible in the event the purified helium header becomes unavailable. The pressure in the helium header will be maintained above reactor pressure. The knock-out pot reduces the probability of moisture in the helium purge header entering the CRD penetrations by trapping any entrained water in the helium. An alarm indicates that water is collecting in the pot.

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BASIS FOR SPECIFICATION LCO 3.1.1/SR 5.1.1 (Continued)

The loss of purge flow to any CRD assembly could result in elevated CRDM temperatures that would require the appropriate monitoring and its associated partial scram testing.

The knock-out pot is approximately 10 inches deep. Verifying that the water level in the knock-out pots is less than 6 inches once every 24 hours ensures that the helium purge flow is not carrying condensed water.

Slack Cable Alarm

The tension of the cables supporting each pair of control rods is monitored by means of slack cable sensing switches (FSAR section 7.2.2.2). A slack cable alarm for a region may indicate a control rod stuck in the guide channels of the core, a parted control rod cable, dropped control rod absorber sections, or a failure of the alarm instrumentation. There are provisions to allow limited motion of the affected control rods (up to 3 inches) to determine whether a rod or cable is stuck or a cable is broken, and various diagnostic techniques can be used to determine the OPERABILITY of the instrumentation.

Actions

The ACTION to initiate a reactor shutdown within 10 minutes if one or more control rod pairs are inoperable due to being immovable (e.g., resulting from excess friction or mechanical interference) is implemented because the cause of the problem may be indicative of a generic control rod pair problem which may affect the ability to safely shut down the reactor. When ACTIONS are to be taken within 10 minutes, no restoration is intended. The ACTION should be taken without delay and in accordance with established procedures.

The ACTIONS providing for continued operation with one control rod pair inoperable due to causes other than being immovable are less restrictive because the SHUTDOWN MARGIN can be met with the highest worth control rod pair fully withdrawn (FSAR Section 3.5.3). Continued operation with CRD motor temperatures greater than 250 degrees F is acceptable provided continued OPERABILITY is demonstrated once per 24 hours by partial scram tests.

BASIS F.. SPECIFICATION LCO 3.1.1/SR 5.1.1 (Continued)

Because the SHUTDOWN MARGIN can be met with the highest worth control rod pair fully withdrawn, an exception to 3.0.4 (which prevents moving up to a higher OPERATIONAL MODE while in an ACTION statement) can be made in this case.

If purge flow is not maintained to two or more CRD penetrations, 2 hours is provided to restore purge flow to the penetrations. This restoration time will provide time to change out a helium bottle or clear any blockage in the subheader, in order to restore purge flow to the CRD penetrations. Degradation of the CRD assembly due to lack of purge flow is a long term effect, and will not occur over a short period of time.

If a slack cable alarm is received, an actual slack cable condition must be confirmed or ruled out within 24 hours. An immovable control rod pair is subject to the SHUTDOWN requirements of Action a and is not considered a slack cable condition in Action i. For the identified slack cable conditions, the affected control rod absorber sections would be inserted into the core or else unaffected as in the case of an instrumentation problem. The consequences of these conditions are conservative, the condition is local, and a 24 hour determination time is acceptable. In the event of an actual slack cable condition, the ACTION to shut down in a controlled manner is acceptable since this is indicative of a local and not generic problem and since determination and resolution will require the removal of that CRDM assembly from the reactor.

Surveillances

The regulating rod pair is the only control rod pair with automatic response capability to a change in flux and is used to offset the negative effects of partial scram tests performed on other control rod pairs. A partial scram test of 2 inches on the regulating rod pair does not induce unacceptable power transients and demonstrates that the control rod pair is moveable.

Performing a partial scram test prior to achieving criticality ensures control rod pair OPERABILITY prior to entering into a higher OPERATIONAL MODE. The full stroke scram test performed during each shutdown is the most accurate method of determining the scram time because the actual scram time is measured over the whole length of the control rod pair versus being extrapolated from a partial distance.

BASIS FOR SPECIFICATION LCO 3.1.1/SR 5.1.1 (Continued)

For control rod pairs that are withdrawn later in the operational schedule, a partial scram test prior to withdrawal is performed to ensure scrammability; no extrapolated scram time is determined. A 10 inch withdrawal from the fully inserted position does not produce consistently meaningful extrapolated scram times due to the CRD mechanism inertia, but it does establish ease of movement. Performing this 10 inch test also minimizes the compensating movements of the regulating rod that would be required for partial scram tests from greater distances.

Performing a full stroke scram test following any CRD maintenance ensures that the OPERABILITY and scram time of the control rod pair was not affected by the maintenance.

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

The specified CHANNEL FUNCTIONAL TEST of the CRD motor temperature and cavity temperature instrumentation will assure that the instrumentation monitoring the CRD temperatures is OPERABLE and capable of detecting any increase in CRD motor temperature

The preventive maintenance program performed on those CRDs replaced each REFUELING CYCLE ensures that by inspecting and replacing as necessary any degraded parts the potential for CRD failure is significantly reduced. Since the regulating rod pair CRD is used more than any other CRD, it will be substituted with another CRD after each REFUELING.

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REACTIVITY CONTROL SYSTEMS

3/4.1.2 CONTROL ROD PAIR POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.1 The position indication instrumentation listed in Table 3.1.2-1 for each control rod pair shall be OPERABLE and capable of determining control rod pair position within 10 inches.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: As shown in Table 3.1.2-1

SURVEILLANCE REQUIREMENTS

4.1.2.1 a. Control rod pair position instrumentation OPERABILITY shall be verified by performing a CHANNEL CHECK on the control rod pair position instrumentation, as follows:

1. Prior to withdrawal from the fully inserted position,

2. Upon full withdrawal, and

3. At least once per 7 days on all fully withdrawn, partially inserted, and fully inserted control rod pairs except for fully inserted control rod pairs incapable of being withdrawn.

   a) During partial scram surveillances on fully withdrawn and partially inserted control rod pairs, the analog rod position indication shall be demonstrated OPERABLE by verifying that the change in analog indication is consistent with the direction of control rod pair travel. The analog and digital position indications must agree within 10 inches of each other. If a larger difference is observed, it shall be assumed that the analog indication is the inoperable channel, unless the analog indication can be proven to be accurate and OPERABLE by another means, and

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SPECIFICATION SR 4.1.2.1 (Continued)

b) During partial scram surveillances on fully withdrawn control rod pairs, the rod-out limit indications shall be demonstrated OPERABLE by verifying that the rod-out indication clears when the control rod pair is inserted less than or equal to 6 inches and is on when the control rod pair is fully withdrawn following the partial scram test.

b. Prior to each reactor startup and the first time during or after startup when the control rod pair is withdrawn from the fully inserted position, the OPERABILITY of the rod-in limit indication shall be verified for each control rod pair by:

1. Verifying that the rod-in limit light is on, when the control rod pair is fully inserted, and

2. Verifying that the rod-in limit light clears, when the control rod pair is withdrawn less than or equal to 6 inches.

c. Prior to each reactor start-up, and during the first outward motion of a control rod pair, the OPERABILITY of the analog and digital position indications shall be verified for each control rod pair by:

1. Verifying that the rod-in limit light is on, when the control rod pair is fully inserted, and

2. Verifying that when the control rod pair is withdrawn a short distance, the rod-in limit light clears, when the analog and digital instrumentation indicates less than 6 inches.

   If the analog and digital position indications indicate 6 or more inches, an engineering evaluation shall be performed to determine the maximum insertion limit for that control rod pair.

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Table 3.1.2-1

CONTROL ROD PAIR POSITION INDICATION SYSTEMS

POSITION OF CONTROL ROD PAIR	POSITION INDICATION INSTRUMENTATION SYSTEMS AVAILABLE	MINIMUM POSITION INDICATION INSTRUMENTATION SYSTEMS OPERABLE	ACTION
Fully Inserted	Rod-in Limit, Independent Means (Watt Meter Test)	1	1, 2
Partially Inserted	Rod-in Limit*, Analog, Digital	2	1, 2
Fully Withdrawn	Rod-out Limit, Analog, Digital	2	1, 2

ACTION STATEMENTS

ACTION 1 With the number of OPERABLE position instrumentation systems less than the Minimum Position Indication Instrumentation Systems OPERABLE requirements, restore the required number of inoperable position indication system(s) to OPERABLE status within 12 hours, or be in SHUTDOWN within the next 12 hours.

ACTION 2 The provisions of Specification 3.0.4 are not applicable for changes between STARTUP, LOW POWER, and POWER. Prior to entry into STARTUP from SHUTDOWN, all the requirements of the LCO must be met, without reliance on the provisions contained in the ACTION statement.

* Demonstrated OPERABLE when last tested.

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BASIS FOR SPECIFICATION LCO 3.1.2.1/SR 4.1.2.1

FSAR Section 7.2.2 assumes a long term misalignment of plus or minus 12 inches on control rod pair position to ensure an acceptable power distribution for core burnup. This allows for a 2 foot separation distance for the control rod pairs of any partially inserted shim group. Assuring a position accuracy of plus or minus 10 inches is consistent with this misalignment allowance and provides for a 4 inch margin for operation when manually adjusting the control rod pairs of the shim group. Each shim control rod pair is normally moved in approximately 2 inch increments during operation to adjust the regulating rod pair to its mid operating position. A 10 inch position accuracy for all control rod pairs is also consistent with a reactivity uncertainty of about 0.003 delta k, which allows for detecting core irregularities, such as an inadvertant release of reserve shutdown material within a single core region. Control rod pair withdrawal procedures require an evaluation if the actual critical control rod pair position differs from the predicted position during initial criticality by this reactivity worth.

Control rod pair position indication system OPERABILITY is required to determine control rod pair positions and to ensure compliance with control rod pair alignment and position requirements of Specifications 3.1.4.1 and 3.1.4.2.

Rod-out and rod-in position indication is provided by cam-actuated switches. The cams are mounted on the same shaft as the rod position potentiometer. The shaft is directly coupled to a cable drum through a gear train and rotates as required for the full length of control rod pair travel. When a control rod pair is withdrawn from the fully inserted position, the limit switch cams release the rod-in switch causing the rod-in light to extinguish. Rod position is transmitted to the console by a potentiometer coupled directly to the drum gearing. The rod-in and rod-out limit switches and the rod position potentiometer transmitters are duplicated to protect against the loss of position indication.

ACTIONS

If analog and rod-in limit indications are OPERABLE but digital and/or rod-out limit indications are inoperable, operation may continue. Since both the analog and digital indications are taken from the same shaft and potentiometer, control rod pair position is still capable of being established with only the analog indication. Rod-in limit indication capability is more critical than rod-out limit indication for the purpose of determining SHUTDOWN MARGIN.

BASIS FOR SPECIFICATION LCO 3.1.2.1/SR 4.1.2.1 (Continued)

If the analog indication is inoperable, operation may continue with one of the following conditions satisfied:

a. If the control rod pair is fully inserted, the position can be established by the rod-in limit indication or verified by an independent means such as the watt-meter test. Since the control rod pair is fully inserted, any other position indication is not required because its position of being fully inserted can be verified and used in the SHUTDOWN MARGIN calculation.

b. For the case when the control rod pair is partially inserted and the digital and rod-in limit indications OPERABLE, control rod pair position can still be established by digital indication and if the control rod pair were to be fully inserted its position could be verified. Rod-in indication OPERABILITY is demonstrated when last tested.

c. For the case with the control rod pair fully withdrawn and rod-out and rod-in limit indications OPERABLE, the control rod pair's position can be established (i.e. fully withdrawn) or if the control rod pair were to be fully inserted, its position could be verified for the SHUTDOWN MARGIN calculation.

If rod-in limit indication were inoperable, operation may continue, because a fully inserted control rod pair's position can be established by an independent means such as the watt-meter test. At a partially inserted or fully withdrawn position, the control rod pair's position can be determined by both digital and analog indications.

If control rod pair position cannot be determined within 12 hours, reactor shutdown is required within the next 12 hours. This ACTION is required to satisfy the control rod pair worth and position requirements of Specification 3.1.4.1, which prevents an unacceptable power distribution.

Surveillances

Control rod pair position indication instrumentation OPERABILITY is verified by performing a CHANNEL CHECK before the control rod pair is withdrawn from the fully inserted position, when it is fully withdrawn, and at least once per 7 days. This surveillance ensures position indication OPERABILITY prior to a reactor startup and during operation.

During the partial scram test (once per 7 days during operation) analog indication is verified OPERABLE by confirming that the change in analog indication is consistent with the direction of control rod pair travel.

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BASIS FOR SPECIFICATION LCO 3.1.2.1/SR 4.1.2.1 (Continued)

If a difference of greater than 10 inches exists between the analog and digital position indications, the analog indication is considered inoperable, unless proven accurate by another means. The analog indication may be proven to be accurate and OPERABLE by fully inserting the control rod pair and verifying that the analog indication is more accurate than the digital indication at the fully inserted position as determined by the rod-in limit indication or the watt-meter test.

The rod-in limit indication is verified to be OPERATING at the fully inserted position when the control rod pair is withdrawn a short distance. This surveillance ensures that a fully inserted control rod pair's position can be established during operation by verifying OPERABILITY of each control rod pair prior to each startup and also when the control rod pair is first withdrawn from the fully inserted position.

To ensure position indication is capable of being established at the partially inserted to fully withdrawn position (during operation) both the analog and digital positions are verified OPERABLE at the fully inserted position when the control rod pair is withdrawn a short distance. This surveillance is performed prior to startup or during the first outward control rod pair motion.

The position indication potentiometers and associated coupling can be damaged by an overtravel of minus 6 inches. This damage is prevented by initially requiring the control rod pair position indication to indicate less than 6 inches when the rod-in limit indication clears and then by procedurally preventing control rod pair insertion past zero, even if rod-in limit indication is not received. The requirement for position indication to be less than 6 inches when rod-in limit indication is received imposes an enhanced accuracy requirement at this position. The result is that since procedurally the control rod pair is not inserted past a zero indication, and if the position indication is within 6 inches of the actual position, then the control rod pair will not be inserted beyond the minus 6 inch damage limit, even if the rod-in limit instrumentation fails. Since control rod pair position instrumentation cannot be recalibrated without removing the CRD from the PCRV, the engineering evaluation provides the necessary procedural controls to establish individual control rod pair insertion limits for control rod pairs whose position indications exceed 6 inches.

3/4.1.2 CONTROL ROD PAIR POSITION INDICATION SYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.2 The position indication instrumentation shall be OPERABLE for each control rod pair capable of being withdrawn and capable of determining control rod pair position within 12 inches with:

a. A rod-out limit indication or analog or digital position indication, when the control rod pair is fully withdrawn, or

b. A rod-in limit indication and either an analog or digital position indication, when the control rod pair is fully inserted.

APPLICABILITY: SHUTDOWN and REFUELING

ACTION: With any of the above required position indication instrumentation inoperable, within 12 hours either:

a. Restore the inoperable position indication instrumentation to OPERABLE status, or

b. Verify full insertion of the control rod pair by other independent means (e.g., watt-meter testing), or

c. Consider the control rod pair fully withdrawn and meet the SHUTDOWN MARGIN requirements of Specification 3.1.3.

SURVEILLANCE REQUIREMENTS

4.1.2.2 a. Control rod pair position instrumentation shall be demonstrated OPERABLE by performing a CHANNEL CHECK as follows:

1. Prior to withdrawal from the fully inserted position,

2. Upon full withdrawal,

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SPECIFICATION SR 4.1.2.2 (Continued)

3. At least once per 7 days on all control rod pairs except for fully inserted control rod pairs which are incapable of being withdrawn, and

4. After an OPERATIONAL MODE change to SHUTDOWN from STARTUP.

The analog and digital position indications shall be within 12 inches of each other. If a larger difference is observed, it shall be assumed that the analog indication is the inoperable channel, unless the analog indication can be proven to be accurate and OPERABLE by another means.

b. During each REFUELING CYCLE, a CHANNEL FUNCTIONAL TEST of each control rod pair's redundant "in" and "out" limit switches and analog and digital rod position indication systems, shall be performed.

c. A CHANNEL CALIBRATION of the control rod pair redundant "in" and "out" limit switches, and the analog and digital rod position indication systems, shall be performed on all CRDs removed for refueling/repair/maintenance.

d. When in REFUELING, prior to each control rod pair withdrawal (unless the surveillance has been performed within the previous 7 days) the OPERABILITY of the analog and digital position indications shall be verified for that control rod pair by:

1. Verifying that the rod-in limit light is on, when the control rod pair is fully inserted, and

2. Verifying that when the control rod pair is withdrawn a short distance, the rod-in limit indication clears when the analog and digital instrumentation indicates less than 6 inches.

If the analog and digital instrumentation indicates 6 or more inches, an engineering evaluation shall be performed to determine the maximum insertion limit for that control rod pair.

BASIS FOR SPECIFICATION LCO 3.1.2.2/SR 4.1.2.2

This specification involves control rod pairs that are either fully inserted or fully withdrawn; therefore, the accuracy requirements are different from those in LCO 3.1.2.1 for operational considerations. The relative reactivity worth for the total control rod pair bank as a function of withdrawal position is given in FSAR Section 3.5.3 (Figure 3.5-2). Experimental results on control rod pair worth versus withdrawal position have indicated a reduced worth for the first portion of control rod pair withdrawal and has been substantiated with new analyses. From this revised calculated data and a calculated bank worth of 0.210 delta k, it can be shown that a reactivity uncertainty of 0.003 delta k results in the total bank position uncertainty of 17 inches at full insertion and 13 inches at full withdrawal. The reactivity uncertainty of 0.003 delta k is acceptable for the SHUTDOWN MARGIN and is consistent with that used to detect core irregularities, such as occasions of inadvertant release of reserve shutdown material within a single core region. Control rod pair withdrawal procedures require an evaluation if the actual critical control rod pair position differs from the predicted position during the approach to criticality by the reactivity worth of 0.003 delta k. Verifying position accuracy within 12 inches is consistent with these control rod pair position uncertainties.

If position indication instrumentation is inoperable, a 12 hour ACTION time is allowed because the SHUTDOWN MARGIN requirements have been met prior to position indication instrumentation inoperability.

Control rod pair position indication instrumentation OPERABILITY is verified by performing a CHANNEL CHECK before the control rod pair is withdrawn from the fully inserted position, when it is fully withdrawn, at least once per 7 days, and after an OPERATIONAL MODE change to SHUTDOWN from STARTUP on those control rod pairs capable of being withdrawn. The Basis for Specification 3.1.4.2 lists the methods of making a control rod pair incapable of being withdrawn. This surveillance ensures position indication OPERABILITY when the reactor is shutdown and during any refueling operations.

During each REFUELING CYCLE a CHANNEL FUNCTIONAL TEST will be performed on the control rod pair redundant "in" and "out" limit switches and the analog and digital rod position indication systems. This surveillance ensures that the entire position indication system is OPERABLE prior to a reactor startup.

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BASIS FOR SPECIFICATION LCO 3.1.2.2/SR 4.1.2.2 (Continued)

In conjunction with CRD removal from the PCRV, a CHANNEL CALIBRATION will be performed on the control rod pair redundant "in" and "out" limit switches, and the analog and digital rod position indication systems. A CHANNEL CALIBRATION on the CRD position indication instrumentation cannot be performed while the control rod pairs are installed in the PCRV; therefore, a calibration is performed only on the control rod pairs removed for refueling/repair/maintenance.

During REFUELING, the rod-in limit indication, and analog and digital indications will be verified OPERABLE (for those control rod pairs capable of being withdrawn) within 7 days prior to control rod pair withdrawal.

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3/4.1.3 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.3 The reactor SHUTDOWN MARGIN shall be greater than or equal to 0.01 delta k.

APPLICABILITY: At all times

ACTION:

a. When in POWER, LOW POWER, and STARTUP, with the SHUTDOWN MARGIN less than 0.01 delta k:

   1. Within 1 hour, insert sufficient control rod pairs to achieve the specified SHUTDOWN MARGIN, or

   2. Be in at least SHUTDOWN within the next 12 hours.

b. When in SHUTDOWN, with the SHUTDOWN MARGIN less than 0.01 delta k, within 1 hour, either:

   1. Insert sufficient control rod pairs to achieve the specified SHUTDOWN MARGIN, or

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

c. When in REFUELING, with the SHUTDOWN MARGIN less than 0.01 delta k:

   1. Immediately suspend all control rod pair or fuel manipulations involving positive reactivity changes, and

   2. Within 1 hour either:

      a) Fully insert sufficient control rod pairs to achieve the specified SHUTDOWN MARGIN, or

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

SURVEILLANCE REQUIREMENTS

4.1.3 SHUTDOWN MARGIN shall be assessed as follows:

a. When in POWER, LOW POWER, or STARTUP:

1. Once per 7 days,

2. In assessing the SHUTDOWN MARGIN the following conditions shall be assumed:

a) The highest worth control rod pair is fully withdrawn,

b) All OPERABLE control rod pairs are fully inserted with all inoperable control rod pairs in their pre-scram position,

c) The CORE AVERAGE TEMPERATURE is equal to 220 degrees F, and

d) Full decay of Xe-135, no buildup of Sm-149, and no decay of Pa-233 beyond that present at shutdown.

b. When in SHUTDOWN:

1. Within 12 hours after each reactor shutdown when all control rod pairs cannot be verified fully inserted,

2. Prior to control rod pair withdrawal, if all control rod pairs are not fully inserted prior to withdrawal action, and

3. Prior to control rod pair withdrawal to achieve criticality, to confirm that upon reaching criticality the SHUTDOWN MARGIN requirement can be met.

SPECIFICATION SR 4.1.3 (Continued)

4. In assessing the SHUTDOWN MARGIN the following conditions shall be assumed:

   a) The highest worth control rod pair is fully withdrawn,

   b) All OPERABLE control rod pairs are fully inserted and inoperable control rod pairs in their known position or fully withdrawn,

   c) The CORE AVERAGE TEMPERATURE is equal to 80 degrees F, and

   d) Full decay of Xe-135, full buildup of Sm-149, and Pa-233 decay as a function of time after shutdown.

c. When in REFUELING:

1. Prior to control rod pair withdrawal, if all control rod pairs are not fully inserted prior to withdrawal action, and

2. Prior to the removal of the control rod pair in a region to be refueled or repaired.

3. In assessing the SHUTDOWN MARGIN the following conditions shall be assumed:

   a) The highest worth control rod pair capable of being withdrawn is fully withdrawn,

   b) Control rod pairs being withdrawn for refueling/repair, SHUTDOWN MARGIN assessment, or OPERABILITY test purposes, are fully withdrawn,

   c) All other OPERABLE control rod pairs are fully inserted and incapable of being withdrawn,

   d) Inoperable control rod pairs are in their known position or fully withdrawn,

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SPECIFICATION SR 4.1.3 (Continued)

e) For planned CORE ALTERATIONS, the core shall be in its most reactive configuration,

f) The CORE AVERAGE TEMPERATURE is equal to 80 degrees F, and

g) Full decay of Xe-135, full buildup of Sm-149, and Pa-233 decay as a function of time after shutdown.

BASIS FOR SPECIFICATION LCO 3.1.3/SR 4.1.3

A. SHUTDOWN MARGIN - OPERATING

The purpose of this LCO is to ensure 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 control 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. 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.

Changes in the isotopic inventory following a reactor shutdown, of fission product poisons Xe-135 and Sm-149, and heavy metal Pa-233, are also considered. These changes are due to the buildup and decay of precursors as well as decay of their current concentration. For Xe-135, both the precursor and Xenon isotope decay in hours, with half-lives of 6 and 9 hours respectively, and consequently Xe-135 initially builds up to a peak value in about 6 hours, and then fully decays in a few days. Since full decay occurs in a few days, it is conservatively assumed to be fully decayed at the time of shutdown. The precursor for Sm-149 has a half-life of a few days, while the decay of Sm-149 occurs over several years, so the buildup occurs over many days and is conservatively assumed to remain at its current value at shutdown. The decay of Pa-233 to fissile U-233 occurs over a period of about 100 days, and it also is assumed to remain at its current value at shutdown.

Any control rod pair that is demonstrated OPERABLE per Specification 3.1.1 will be assumed to be fully inserted and any inoperable control rod pair will be assumed to be at its pre-scram position. This is consistent with FSAR Section 3.5.3, which demonstrates that there is at least 0.014 delta k SHUTDOWN MARGIN with one control rod pair fully withdrawn under any core condition in the equilibrium core and larger for any core condition prior to the equilibrium core.

Assessment of the SHUTDOWN MARGIN requirements at least once per 7 days ensures that changes in the core reactivity as a result of burnup have not occurred which would make the previous verification invalid. The core reactivity changes as a result of burnup occur slowly and a 7 day surveillance during operation is sufficient. In addition, the ACTION statements of Specification 3.1.1 require more frequent assessment if a control rod pair is determined inoperable, or its exact position is uncertain.

BASIS FOR SPECIFICATION LCO 3.1.3/SR 4.1.3 (Continued)

B. SHUTDOWN and REFUELING

The purpose of this specification is to ensure that during SHUTDOWN and REFUELING 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 some 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 12 hour delay in verification of the SHUTDOWN MARGIN is sufficient for the purpose of this specification.

This specification need only require that the control rod pair be actually inserted to achieve the specified SHUTDOWN MARGIN. Since full insertion can be verified by either rod position indication or another independent means, such as watt-meter testing per Specification 3.1.2.2, some additional time has been allowed.

The Reserve Shutdown System was provided to ensure shutdown even in the event of failure to insert control rod pairs. It is adequate to ensure shutdown even if all control rod pairs fail to insert (FSAR Section 3.5.3). However, the contribution to the SHUTDOWN MARGIN by the addition of reserve shutdown material into a core region already containing an inserted control rod pair is minimal. Therefore, it is sufficient to activate the reserve shutdown material only in those regions whose control rod pairs are not fully inserted.

For SHUTDOWN, the specified SHUTDOWN MARGIN assumes the full withdrawal of the highest worth control rod pair. For REFUELING, (which can include either fuel or control rod pair manipulations) since all control rod pairs are disabled, except those involved with REFUELING per Specification 3.1.4.2, the requirement includes the addition of the highest worth control rod pair capable of being withdrawn in the SHUTDOWN MARGIN calculation. Disabling of control rod drives by disabling the electrical supply to the drive motors or placing the reactor mode switch in the "off" position results in the inability to withdraw the control rod pair by action of the drive motor. Therefore, the accidental withdrawal of any control rod pair in this manner does not have to be assumed in the SHUTDOWN MARGIN calculation.

The ACTION statement of Specification 3.1.4.2, Control Rod Pair Position Requirements-SHUTDOWN, requires completion of the assessment of the SHUTDOWN MARGIN within 12 hours.

BASIS FOR SPECIFICATION LCO 3.1.3/SR 4.1.3 (Continued)

Within the first 24 hours after shutdown, the SHUTDOWN MARGIN is significantly larger than specified due to higher core temperatures and the presence of Xe-135 and Pa-233. A 12 hour delay will not compromise the validity of this specification.

Assessment of the SHUTDOWN MARGIN prior to any control rod pair withdrawal, if all control rod pairs are not fully inserted, prior to withdrawal to achieve criticality, and prior to removal of a control rod pair for refueling/repair, ensures that the requirements of this specification will be met during these ACTIONS.

REACTIVITY CONTROL

3/4.1.4 CONTROL ROD PAIR POSITION AND WORTH REQUIREMENTS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.4.1 Control rod pair position and worth requirements shall be as follows:

a. Control rod pairs (except the regulating rod pair) shall be withdrawn or inserted in groups (three control rod pairs per group) except during scrams, control rod pair runbacks, partial scram surveillance, or manipulations of additional control rod pairs permitted by Specification b.2 below.

b. All control rod pairs shall be either fully inserted or fully withdrawn except during partial scram testing and:

1. One shim group and the regulating rod pair may be in any position, and

2. Up to six additional control rod pairs may be inserted up to 2 feet.

c. The maximum calculated control rod pair worth shall not exceed:

1. 0.047 delta k, with the reactor critical at approximately 1.0 E-07 percent RATED THERMAL POWER (source power), and

2. At full power, that worth which would result in Rod Withdrawal Accident (RWA) consequences equal to those described for the worst case RWA in the AEC Safety Evaluation of Fort St. Vrain dated January 20, 1972.

APPLICABILITY: POWER*, LOW POWER, and STARTUP

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* See Special Test Exceptions Specification 3.10.1.

SPECIFICATION LCO 3.1.4.1 (Continued)

ACTION:

a. With any control rod pair or group not in compliance with its position requirements either:

1. Restore the control rod pair(s) to an acceptable configuration within 4 hours, or

2. Be in at least LOW POWER within the next 12 hours, and SHUTDOWN within the following 12 hours.

b. With any control rod pair not in compliance with its worth limits, be in at least SHUTDOWN within 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.4.1.1 At least once every 12 hours, each control rod pair position shall be verified to be in compliance with the above requirements.

4.1.4.1.2 At the beginning of each REFUELING CYCLE, the reactivity worth of the control rod pair groups withdrawn from LOW POWER to POWER in the withdrawal sequence, shall be measured. The measured group worths shall be compared with the calculated group worths to verify that the calculated criteria upon which the selection of the control rod pair sequence was 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.

BASIS FOR SPECIFICATION LCO 3.1.4.1/SR 4.1.4.1.1, 4.1.4.1.2

The specification of a control rod pair withdrawal sequence (Specification 5.3.4, Reload Segment Design) and position requirements during STARTUP and LOW POWER operation is required to:

a. Assist in evaluating the reactivity worth of control rod pairs withdrawn during the approach to criticality 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 rod pairs are still inserted, and

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

The specification of a control rod pair withdrawal sequence and position requirements during POWER are required to yield an acceptable power distribution. In addition, the sequence ensures that the combination of maximum single control rod pair worth and available core temperature coefficients, in the event of an accidental control rod pair withdrawal, will result in a transient with consequences less severe than those analyzed in the FSAR. The RWA analyzed in the FSAR is consistent with the RWA evaluation in the AEC Safety Evaluation of Fort St. Vrain dated January 20, 1972.

The maximum calculated control rod pair worth limit of 0.047 delta k at approximately 1.0 E-7 percent power is based on the Maximum Worth Control Rod Pair Withdrawal at Source Power analysis in FSAR Section 14.2.2.7.

The RWA analysis at rated power, as described in the FSAR, is based on a maximum control 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 0 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 control rod pair worth. In addition, since the temperature coefficients are greater at the beginning of the cycle, a single control rod pair worth as much as 0.016 delta k is acceptable, i.e., the consequences of an RWA are less severe (FSAR Section 14.2.1). A value larger than 0.012 delta k for a single control rod pair can be safely accommodated if fuel temperatures are lower than 1500

BASIS FOR SPECIFICATION LCO 3.1.4.1/SR 4.1.4.1.1, SR 4.1.4.1.2 (Continued)

degrees F and/or the temperature defect between refueling temperature (220 degrees F) and operating temperature (1500 degrees F) is greater than 0.028 delta k (FSAR Section 14.2.1.1).

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 pair withdrawal sequence for each REFUELING CYCLE which has peaking factors within these power peaking factor limits ensures that the criteria upon which Specification 2.1 is based, are met.

The presence of too many partially inserted control 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 control rod pair withdrawal sequence for each REFUELING CYCLE will be maintained during normal operation if the control rod pairs are inserted and withdrawn in sequence and if partially inserted control rod pairs are limited as noted above (FSAR Section 3.5.4).

The six additional control rod pairs which may be inserted up to 2 feet into the core will permit the operator to move control rod pairs to assist in regulating the core region outlet temperatures to those specified in Specification 3.1.2. 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%.

The runback function inserts two pre-selected groups of three control rod pairs during rapid load reduction (FSAR Section 7.2.1.2). The partial insertion of these control rod pairs, (FSAR Section 3.5.4.2) 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 kernel migration (Specification 2.1) would occur with this condition in the core for up to 4 hours.

The ACTION to be in at least LOW POWER within the next 12 hours and SHUTDOWN in the following 12 hours requires an orderly shutdown to reduce plant load and temperatures in a controlled manner. Core temperatures are significantly reduced as lower power levels are reached, and in STARTUP negligible fuel particle kernel migration would occur as long as the minimum helium flow requirements (Specification 3.2.4) are maintained.

BASIS FOR SPECIFICATION LCO 3.1.4.1/SR 4.1.4.1.1, SR 4.1.4.1.2 (Continued)

Verification of control rod pair positions (by monitoring position indication) once per 12 hours is consistent with the verification of INDIVIDUAL REFUELING REGION OUTLET TEMPERATURES once per 12 hours (Specification 4.2.2.1).

The measurement of control rod pair 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 pair group worths in the core configuration for that cycle. The criteria used in selecting the control rod pair sequence is based on calculated data for the maximum worth for any individual control rod pair as well as the calculated peaking factors (region, intra-region, and axial) in the normal operating control rod pair 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 pair worths in that specific configuration.

The acceptance criteria for the comparison of measured versus calculated control rod pair 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 pair 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 pair worth calculations was developed for seven column regions. In addition, since the control rod pairs are located in the central column and this column for a five column region is immediately adjacent to the reflector, their reactivity worth is substantially less than the other control rod pair groups. These groups are typically worth less than 0.010 delta k. Because of the low worth and the analytical uncertainty, a larger range for the acceptance criteria is required.

REACTIVITY CONTROL SYSTEM

3/4.1.4 CONTROL ROD PAIR POSITION REQUIREMENTS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.4.2 All control rod pairs shall be fully inserted and incapable of being withdrawn except:*

1. Up to two control rod pairs may be removed from the PCRV, and

2. Additional control rod pairs may be withdrawn for SHUTDOWN MARGIN assessment or OPERABILITY tests.

APPLICABILITY: SHUTDOWN AND REFUELING

ACTION: a. Within 1 hour after each reactor shutdown, if more than two control rod pairs are not verified to be fully inserted, either:

1. Insert at least all but two control rod pairs to the fully inserted position, or

2. Insert reserve shutdown material in at least those regions where control rod pairs are not verified to be fully inserted, beyond the allowable two.

b. Subsequent to 1 hour after reactor shutdown, with less than the above requirements:

1. Immediately suspend all operations involving CORE ALTERATIONS, control rod pair movements resulting in positive reactivity changes or movement of IRRADIATED FUEL.

2. Within 12 hours either:

a) Insert any control rod pair capable of being inserted and verify the SHUTDOWN MARGIN requirements of Specification 3.1.3 are met, or

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

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* The SHUTDOWN MARGIN requirements of Specification 3.1.3 (for SHUTDOWN and REFUELING) shall be maintained for all these control rod pair configurations.

SURVEILLANCE REQUIREMENTS

4.1.4.2 a. Control rod pair positions for all control rod pairs capable of being withdrawn shall be monitored for compliance with Specification 3.1.4.2.a above, at least once every 12 hours.

b. Following each reactor shutdown, each control rod pair shall be verified to be at the fully inserted position by:

1. The rod-in position indication, or

2. The use of an independent control rod pair position verification method (e.g., watt-meter test).

Control rod pairs known to be fully inserted prior to the shutdown may be excluded from the above verifications.

c. Prior to the removal of more than one control rod drive assembly from the PCRV, the SHUTDOWN MARGIN shall be explicitly calculated per the assumptions specified in SR 4.1.3.

d. Upon full withdrawal of a control rod pair selected for removal from the PCRV, and prior to disabling its scram capabilities, the SHUTDOWN MARGIN shall be assessed by withdrawing one or more additional control rod pairs with a calculated worth greater than or equal to 0.01 delta k, plus any calculated positive worth of the planned CORE ALTERATION, verifying subcriticality, and then reinserting.

BASIS FOR SPECIFICATION LCO 3.1.4.2/SR 4.1.4.2

This specification ensures that a sufficient number of control rod pairs are fully inserted to keep the reactor in a shutdown condition (SHUTDOWN MARGIN greater than or equal to 0.01 delta k) in SHUTDOWN and REFUELING.

Prior to refueling a region, the control rod pair in that region and the control rod pair in the region next in sequence to be refueled will be withdrawn. Additional predesignated control rod pairs will also be withdrawn and subcriticality will be verified. The calculated minimum reactivity worth of the additional predesignated control rod pairs is 0.01 delta k plus the reactivity difference between the new and spent fuel in the region to be refueled, plus the temperature defect between the refueling temperature and 80 degrees F. After subcriticality has been verified, the predesignated control rod pairs will be fully reinserted. Withdrawal of the predesignated control rod pairs ensures a SHUTDOWN MARGIN of greater than or equal to 0.01 delta k at 80 degrees F with new fuel loaded into the refueled region.

Making all of the fully inserted control rod pairs incapable of being withdrawn by placing the reactor mode switch in the "off" position or [illegible] the electrical supply to the motors, ensures that [illegible] analyzed core configuration which might result in criticality will not exist.

A SHUTDOWN MARGIN of greater than or equal to 0.01 delta k after reactor shutdown (automatic scram or controlled) is ensured by the 1 hour ACTION to either insert all but two control rod pairs or insert reserve shutdown material in those regions where control rod pairs cannot be verified to be fully inserted, beyond the allowable two. Control rod pairs may be verified fully inserted by rod-in indication (either rod-in limit, analog, or digital position indication), or by other independent means (e.g., watt-meter test) as time permits. The 12 inch limit ensures reactivity control, as discussed in the Bases for LCO 3.1.2.2. Experience has shown that a control rod pair which is not fully inserted by a scram may still be fully inserted manually with its control rod drive motor. If the control rod pair cannot be fully inserted with its drive motor, reserve shutdown material will be inserted into that region, ensuring a SHUTDOWN MARGIN of greater than or equal to 0.01 delta k.

BASIS FOR SPECIFICATION LCO 3.1.4.2/SR 4.1.4.2 (Continued)

If any requirements of the LCO are not met while the reactor is in SHUTDOWN or REFUELING, any control rod pair or fuel manipulations which would result in a positive reactivity addition will be suspended immediately and within 12 hours any withdrawn control rod pairs will be fully inserted. If a SHUTDOWN MARGIN of greater than or equal to 0.01 delta k is not met by fully inserting the control rod pairs, sufficient reserve shutdown material will be inserted to achieve the specified SHUTDOWN MARGIN. This ACTION ensures a SHUTDOWN MARGIN of greater than or equal to 0.01 delta k during reactor shutdown or refueling operations.

The reserve shutdown material provides an effective method of reactivity control when inserted into core regions where the control rod pairs have not been fully inserted. Because of the proximity to the control rod pairs, it has almost no additional worth when inserted in regions where the control rod pairs are inserted. Therefore, to ensure an adequate SHUTDOWN MARGIN, it need only be inserted into those core regions where full insertion of the control rod pairs cannot be demonstrated.

Verifying control rod pair positions once every 12 hours, ensures that control rod pair position can be monitored during control rod pair manipulations performed in refueling operations.

After each shutdown, verifying that each control rod pair is fully inserted ensures that the position of each control rod pair is known and that the SHUTDOWN MARGIN assessment is accurate.

Demonstrating that a SHUTDOWN MARGIN of greater than or equal to 0.01 delta k exists prior to removing more than one control rod drive assembly from the PCRV ensures that criticality will not be achieved and the SHUTDOWN MARGIN requirements will be maintained after the control rod pair is removed.

REACTIVITY CONTROL SYSTEMS

3/4.1.5 REACTIVITY CHANGE WITH TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.1.5 The reactivity change due to a CORE AVERAGE TEMPERATURE increase between 220 degrees F and 1500 degrees F, shall be at least as negative as 0.031 delta k, but no more negative than 0.065 delta k throughout the REFUELING CYCLE.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: With the reactivity change outside of the above limits, be in SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 At the beginning of each REFUELING CYCLE the reactivity change as a function of CORE AVERAGE TEMPERATURE change (temperature coefficient) shall be measured and integrated to verify that the measured reactivity change is within the above limits.

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 actuation of the reserve shutdown system, peak fuel temperatures will be sensitive to the magnitude of the negative temperature coefficient.

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 degree F temperatures associated with the nominal RATED THERMAL POWER value, ensures temperature coefficients at least as negative as those used in the FSAR accident analysis. All control rod pair withdrawal transients assume 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 reactivity temperature defect of 0.065 delta k (0.072 delta k minus 0.007 delta k for uncertainty) assures that there is sufficient reactivity control to ensure reactor SHUTDOWN in the unlikely event that all control rod pairs cannot be inserted and the reserve shutdown system has been actuated.

The reactivity worth of the reserve shutdown system was calculated to be 0.130 delta k in the equilibrium core (FSAR Section 3.5.3). From calculated excess reactivity data in Table 3.5-4 and Section 3.5.3 of the FSAR it is seen that the maximum excess reactivity in the equilibrium core with the CORE AVERAGE TEMPERATURE of 220 degrees F, Xe-135 decayed, Sm-149 built up, and 2 weeks Pa-233 decay, is 0.102 delta k. Assuming no control rods are inserted and the reserve shutdown system has been activated, the excess SHUTDOWN MARGIN for that excess reactivity is 0.028 delta k, (0.130 delta k minus 0.102 delta k). The calculated reactivity temperature defect for that cycle is 0.044 delta k. Therefore, if the reactivity temperature defect were as large as 0.072 delta k (0.044 delta k plus 0.028 delta k) reactor SHUTDOWN could be ensured for at least 2 weeks even for the unlikely event that all control rods failed to insert, and the reserve shutdown system was actuated.

BASIS FOR SPECIFICATION LCO 3.1.5/SR 4.1.5 (Continued)

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 ensure 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.

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. 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 of the minimum reactivity temperature defect cannot be made. However, by comparing the calculated value at the beginning of the cycle with the measured value, an evaluation for compliance can be made using the calculated value at the end of cycle. Performance of the Surveillance Requirement verifies the assumptions used in the analysis.

REACTIVITY CONTROL SYSTEMS

3/4.1.6 RESERVE SHUTDOWN SYSTEM - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.6.1 All reserve shutdown (RSD) units shall be OPERABLE with:

a. At least 1500 psig pressure in their individual helium gas bottle supplies, and

b. At least 500 psig pressure in the Alternate Cooling Method (ACM) nitrogen bottles which provide a backup means of actuating the RSD hopper pressurization valves.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: a. With one RSD unit inoperable, operation may continue provided that an OPERABLE spare RSD unit is available.

b. With two or more RSD units inoperable, restore all but one inoperable RSD unit to OPERABLE within 24 hours, or be in at least SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable for changes between STARTUP, LOW POWER, and POWER. Prior to entry into STARTUP from SHUTDOWN, all the requirements of this LCO must be met, without reliance on the provision: contained in the ACTION statements.

SURVEILLANCE REQUIREMENTS

4.1.6.1 The reserve shutdown system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the pressure of each helium gas bottle is at least 1500 psig.

b. At least once per 7 days by verifying that the pressure of each ACM nitrogen bottle is at least 500 psig.

SPECIFICATION SR 4.1.6.1 (Continued)

c. At least once per 92 days by:

1. Pressurizing each of the 37 RSD hoppers above reactor pressure, as indicated by operation of the hopper pressure switch,

2. Operating the ACM quick disconnect couplings, and

3. Performing a CHANNEL FUNCTIONAL TEST of the instrumentation which alarms at low pressure in the RSD actuating pressure lines.

d. At least once per 366 days by performing a CHANNEL CALIBRATION of the gas pressure instrumentation.

e. Following entry of condensed water into any RSD system hopper(s) (see Specification 3.1.1 ACTION h), by performing the Surveillance requirements identified in SR 4.1.6.2.d.4.

3/4.1.6 RESERVE SHUTDOWN SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.6.2 Reserve shutdown (RSD) units on control rod drive assemblies for which control rod pairs are capable of being withdrawn shall be OPERABLE (except RSD units in any control rod drive assemblies removed for refueling/repair) with:

a. At least 1500 psig pressure in their individual Helium gas bottle supplies, and

b. At least 500 psig pressure in the Alternate Cooling Method (ACM) nitrogen bottles which provide a backup means of actuating the RSD hopper pressurization valves.

APPLICABILITY: SHUTDOWN and REFUELING

ACTION: With less than the above required RSD units OPERABLE, within 24 hours either:

a. Return all control rod pairs (except the ones removed for refueling/repair) to the fully inserted position, or

b. Ensure SHUTDOWN MARGIN requirements are met (Specification 3.1.3), or

c. Insert sufficient RSD material to maintain SHUTDOWN MARGIN requirements.

SURVEILLANCE REQUIREMENTS

4.1.6.2 The reserve shutdown system shall be demonstrated OPERABLE:

a. Prior to withdrawal and at least once per 7 days thereafter, for those control rod pairs capable of being withdrawn, by verifying that the pressure of each required individual hopper He gas bottle is at least 1500 psig.

b. Prior to withdrawal and at least once per 7 days thereafter, for those control rod pairs capable of being withdrawn, by verifying that the pressure of each required ACM nitrogen bottles is at least 500 psig.

SPECIFICATION SR 4.1.6.2 (Continued)

c. At least once per 366 days by performing a CHANNEL CALIBRATION of the gas pressure instrumentation.

d. At each REFUELING outage by:

1. Demonstrating that each subsystem is OPERABLE by actuating each group of pressurizing valves from the control room and verifying that the valves open. The capability of pressurizing the corresponding hoppers need not be demonstrated during this test.

2. Performing a CHANNEL CALIBRATION of the RSD hopper pressure switches at the time of control rod drive preventive maintenance (Specification 4.1.1).

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), and verifying that there is no deformation or corrosion that could affect RSD system OPERABILITY.

4. Functionally testing two RSD assemblies, removed from the core during the current refueling, out of the core. One assembly shall contain 20 weight percent boronated material and the other 40 weight percent boronated material. The tests consist of pressurizing the RSD hopper to the point of rupturing the disc and releasing the RSD material.

The RSD material from the tested RSD 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 RSD 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.

e. Following entry of condensed water into any RSD system hopper(s) (see Specification 3.1.1 ACTION h.), by performing the Surveillance Requirements identified in SR 4.1.6.2.d.4.

BASIS FOR SPECIFICATION LCO 3.1.6/SR 4.1.6

The reserve shutdown (RSD) system must be capable of achieving reactor shutdown in the event that the control rod pairs fail to insert.

After extended power operation, the RSD system must add sufficient negative reactivity to overcome the temperature defect between 1500 and 220 degrees F, the decay of Xe-135, and some decay of Pa-233 to U-233. The buildup of Sm-149 also adds negative reactivity and is taken into account in reactivity evaluations.

The core reactivity increase due to core cooldown and Xe-135 decay occurs within a few days and was calculated to be between 0.089 delta k and 0.081 delta k, at the beginning and end of the initial cycle, respectively, and about 0.076 delta k for the mid cycle of the equilibrium core. The reactivity increase is largest in the initial core where the thorium loading is high and decreases through the first six cycles to a minimum value for the equilibrium core. The reactivity increase due to Sm-149 buildup and Pa-233 decay occurs over several weeks to months and increases the core excess reactivity for the equilibrium core by about 0.007 delta k during the first 14 days, and by about 0.024 delta k after a few months, including full Pa-233 decay. Therefore, the reactivity control requirement for the RSD system, including an allowance of 0.01 delta k for SHUTDOWN MARGIN, in the absence of any control rod pairs being inserted is 0.098 delta k for the initial core and 0.093 delta k for the equilibrium core after 14 days of Pa-233 decay and 0.121 delta k and 0.110 delta k after full Pa-233 decay. (FSAR Section 3.5.3).

The calculated worth for the RSD system as noted in FSAR Section 3.5.3 is at least 0.14 delta k in the initial core, and 0.13 delta k in the equilibrium core. The worth of the RSD System with the maximum worth RSD unit inoperable for those cases is at least 0.12 delta k in the initial core and 0.11 delta k in the equilibrium core, which is sufficient to ensure SHUTDOWN during the first 14 days of Pa-233 decay.

Generally, inoperable RSD units are capable of being restored to OPERABLE status within 24 hours. However, in the unlikely event that an inoperable RSD unit cannot be restored to OPERABLE within this time, there is adequate time (at least 14 days due to the slow Pa-233 decay as discussed in the BASIS for Specification 3.1.3) following a shutdown using the RSD system, to allow for corrective action of changing out a CRD assembly. A spare RSD unit is considered available if it is on site.

BASIS FOR SPECIFICATION LCO 3.1.6/SR 4.1.6 (Continued)

Ensuring SHUTDOWN MARGIN requirements for a CORE AVERAGE TEMPERATURE greater than or equal to 220 degrees F is acceptable and provides for changing out a Control Rod Drive (CRD) assembly, if necessary. 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.

Two or more RSD units may be inoperable for 24 hours to provide a reasonable time for repair. This is permissible because the control rod pairs are available to shut down the reactor in the unlikely event that a shutdown would be required during this short period of time.

A minimum pressure of 1500 psig in the individual helium gas bottle supplies is adequate because the pressure required to burst the rupture discs is 1100 psig (FSAR Section 3.8.3). The rupture discs are designed and have been tested to burst at a differential pressure of 165 plus or minus 50 psi.

A minimum pressure of 500 psig in the ACM nitrogen bottles is adequate because the required set pressure is 220 psig. A set pressure of 220 psig is based on stroking a bank of 10 RSD valves one time and keeping the regulator fully open. This value also compensates for minor line losses and system leakages.

Each of the 37 RSD hoppers shall be pressurized above reactor pressure at least once per 92 days. Two redundant pressurizing valves will be opened using local test switches and the corresponding hopper pressure observed to increase. To prevent releasing absorber material, the high pressure gas cylinder is isolated and the pressurized actuating line is vented prior to the test. Pressurization is accomplished using test gas at a pressure differential of approximately 40-70 psi above reactor pressure, which is below the 115 psi differential pressure required to rupture the disc. The hopper pressure should increase at least 10 psi above reactor pressure, as indicated by the hopper high pressure alarm.

A CHANNEL FUNCTIONAL TEST will be performed on the low pressure alarm instrumentation at least once per 92 days to ensure that the minimum require rupture gas pressure can be monitored.

A CHANNEL CALIBRATION will be performed on the gas pressure instrumentation at least once per 365 days to ensure reliable monitoring of the helium and nitrogen gas supplies.

BASIS FOR SPECIFICATION LCO 3.1.6/SR 4.1.6 (Continued)

In the event that condensed water enters into any RSD system hoppers, (and during each refueling outage) two RSD hoppers shall be functionally tested out of the core. One assembly will contain 20 weight percent and the other 40 weight percent boronated material. The RSD hopper will be pressurized to the point of rupturing the disc and releasing the poison material. The material will be visually examined for boric acid crystallization and chemically analyzed for boron carbide and leachable boron content.

Specification 3.1.6.2, RSD hoppers in the SHUTDOWN and REFUELING MODES, only requires RSD units to be OPERABLE for those control rod pairs capable of being withdrawn because the worth of the control rod pair(s) removed from the PCRV has been accounted for in the SHUTDOWN MARGIN and the worth of the RSD material in regions whose control rod pair are inserted adds little to the SHUTDOWN MARGIN.

The ACTION time of 24 hours is adequate because the reactor has already been shutdown and the SHUTDOWN MARGIN requirements met, versus verifying SHUTDOWN MARGIN requirements immediately after a shutdown.

At each refueling, each group of pressurizing valves will be actuated from the control room to verify that the valves open.

At each refueling, the RSD hopper pressure switches which measure the pressure differential between the hoppers and the reactor will be calibrated as individual control and orifice assemblies are removed from the reactor for servicing and maintenance. These switches alarm high pressure for pressurization testing or actual system operation.

The refueling penetration pipe sections will be visually examined for deformation and corrosion following disassembly for refueling or maintenance.

REACTIVITY CONTROL SYSTEMS

3/4.1.7 REACTIVITY STATUS

LIMITING CONDITION FOR OPERATION

3.1.7 The difference between the observed and expected core reactivity, shall not exceed 0.01 delta k.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: With a core reactivity difference greater than the above limit, be in SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.7 a. At each startup upon reaching initial criticality, upon reaching 100% of RATED THERMAL POWER, and at least once per 7 days while in POWER, the reactivity status of the core shall be determined and compared with expected reactivity to ensure that the above limit is satisfied. The expected reactivity is based upon current core burnup at full power, and therefore reactivity must be normalized to a calculated reactivity for that core burnup.

b. The requirements of Specification 4.0.4 are not applicable.

BASIS FOR SPECIFICATION LCO 3.1.7/SR 4.1.7

The calculated base reactivity for the core for each fuel cycle is obtained from a nuclear analysis assuming a core condition of all control rod pairs inserted, an average temperature of 80 degrees F, no Xe-135 or Sm-149, and all Pa-233 decayed to U-233. The reactivity worth for each of these individual core components, namely temperature, fission product concentration, burnup, etc., is also explicitly calculated as a function of core condition.

The observed core reactivity is that determined from the calculated core data, the observed core condition, and the observed critical control rod configuration. The expected reactivity is determined from the observed core condition and the calculated reactivity as a function of core burnup. The comparison is between the observed critical control rod configuration and the expected critical control rod configuration for the observed core condition.

An unexpected and/or unexplained change in the observed core 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 shall 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 control rod pair fully withdrawn is always maintained (Specification 3.1.3).

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 calculated base reactivity curve for use with each cycle as well as any changes to this data during the cycle will be approved by the Nuclear Facility Safety Committee (NFSC) prior to use (Specification 6.5.2.9). This will ensure that the cause for all calculated changes is adequately understood and controlled.

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

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CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

3/4.2.1 CORE IRRADIATION

LIMITING CONDITION FOR OPERATION

3.2.1 The maximum in-core irradiation of the fuel elements, control rods, and reflector elements immediately adjacent to the active core shall not exceed the equivalent of 1800 Effective Full Power Days (EFPDs).

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: With the in-core irradiation lifetime of the fuel elements, control rods, or reflector elements adjacent to the active core exceeding the above limit be in SHUTDOWN within 72 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 Prior to entering STARTUP following each refueling, it shall be determined that the in-core irradiation lifetime of the fuel elements, control rods, and reflector elements adjacent to the active core will be less than the above limit for the duration of the next cycle.

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BASIS FOR SPECIFICATION LCO 3.2.1/SR 4.2.1

The integrity of the fuel particle coatings and graphite dimensional changes are 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 to 1800 EFPDs, in conjunction with the peaking factor limits of Specification 5.3, will ensure that the coated fuel particles and graphite will remain within the demonstrated irradiation test values. The burnup and irradiation test results (FSAR Appendix A.2) are generally described in terms of percent Fissions per Initial Metal Atom (FIMA) for both the fissile and fertile particles.

The basis for the design lifetime for the fuel elements, control rods, and replaceable reflector elements is described in Sections 3.2 and 3.8 of the FSAR. For the fuel and reflector elements (FSAR Section 3.2.2.2), consideration is given to mechanical loads and stresses during both steady state and transient operation. The elements' structural integrity will be sufficient to permit safe removal from the core after 1800 EFPDs of operation.

For the control rods (FSAR Sections 3.2.2.6 and 3.8.1.2), this lifetime will ensure that reactivity control is maintained even if the control rods were inserted for the total duration, without any significant loss of absorber worth or structural or functional deterioration.

The core irradiation limit of 1800 EFPDs is related to a residence time in the core for each element. An evaluation of the residence time records and an allowance for the duration of the next fuel cycle will ensure that this specification is not exceeded during the next cycle of operation for any element.

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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:

a) Any of the nine regions whose inlet orifice valves are most fully closed, and

b) Any region with control rods inserted more than 2 feet.

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

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, LOW POWER, and STARTUP

ACTION:

a. With an INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE exceeding the above limits by less than 50 degrees F, restore the out-of-limit condition within 24 hours, or be in SHUTDOWN within the next 12 hours.

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SPECIFICATION LCO 3.2.2 (Continued)

b. With an INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE exceeding the above limits by greater than or equal to 50 degrees F, but less than 100 degrees F, restore the out-of-limit condition within 2 hours, or be in SHUTDOWN within the next 12 hours.

c. With an INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE exceeding the above limits by greater than or equal to 100 degrees F, immediately initiate a reduction in THERMAL POWER and restore the out-of-limit condition, or be in SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 All INDIVIDUAL REFUELING REGION OUTLET TEMPERATURES shall be determined to be within the above limits at least once per 12 hours, by obtaining the INDIVIDUAL REFUELING REGION OUTLET TEMPERATURES as determined from either the region outlet temperature thermocouples or from the calculated value using the COMPARISON REGION.

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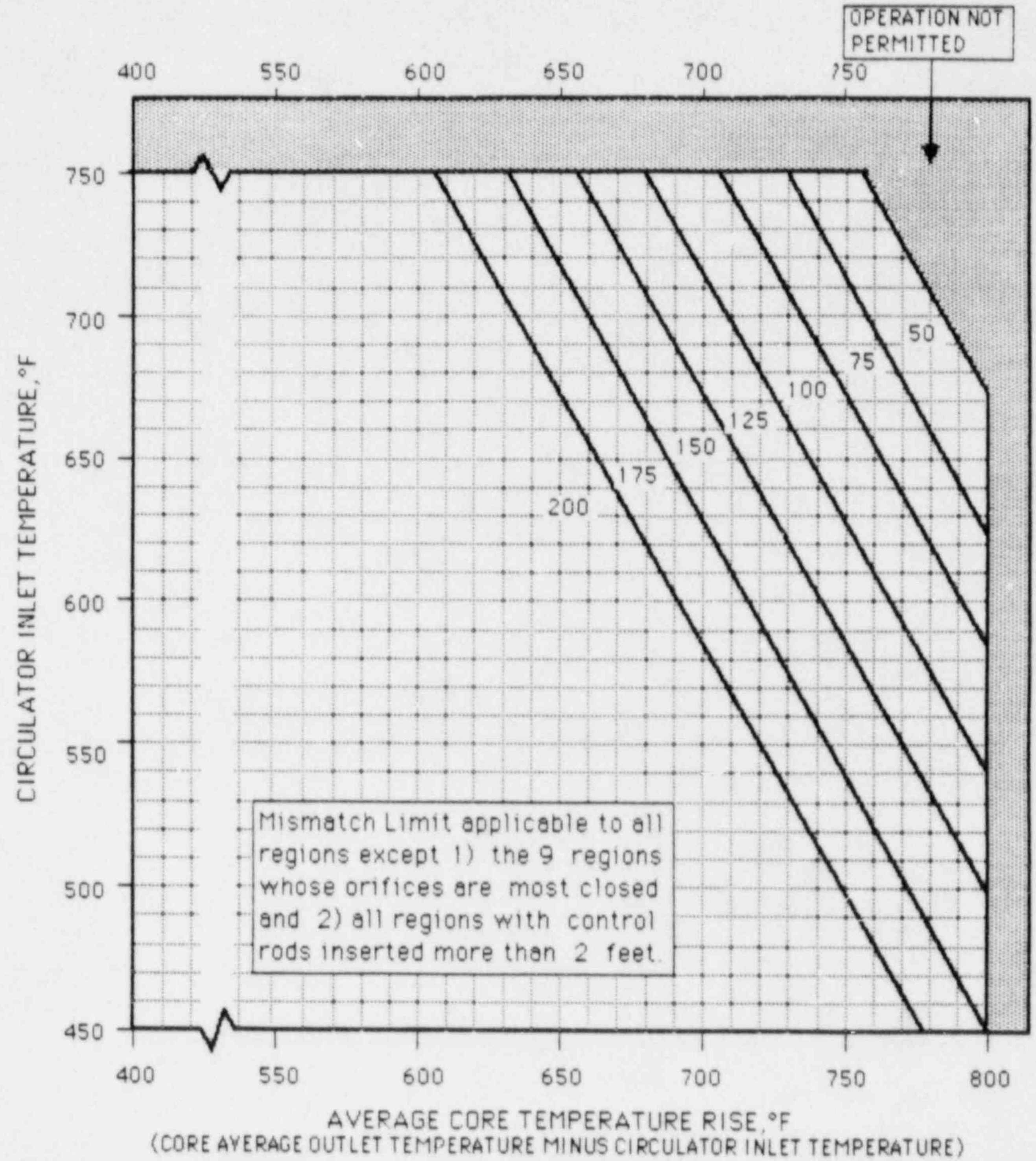

ALLOWABLE DIFFERENCE (MISMATCH LIMIT) BETWEEN INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE AND CORE AVERAGE OUTLET TEMPERATURE

Figure 3.2.2-1

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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, Specification 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 values stated in the FSAR 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 ensure that Specification 2.1.1 remains valid and that the integrity of the fuel particles is preserved.

The ACTION times provided for corrective action when temperatures exceed the limits of the specification represent conditions significantly below the core SAFETY LIMIT determination, Specification 2.1.1.

A COMPARISON REGION is a core refueling region whose power, flow rate, 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 TEMPERATURES. 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 temperatures. (FSAR Section 3.6.6.3)

BASIS FOR SPECIFICATION LCO 3.2.2/SR 4.2.2 (Continued)

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 rate 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 indicated that, by the 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 any of Regions 20 and 32 through 37 is higher than that based upon the COMPARISON REGION conditions, the measured INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE is assumed to be correct.

The region outlet temperatures are dependent on power density and coolant flow rate within each refueling region. The region power density is dependent primarily on the control rod configuration, and the coolant flow rate within each region is controlled by the core inlet orifice valve position. The control rod and orifice valve position configurations are determined by the reactor power level rather than by variations in fission product poisons or core burnup, which are slowly varying. Therefore, monitoring the region outlet temperatures once per shift (12 hours) for changes in reactor power will ensure that the temperature limits in Specification 3.2.2 can be satisfied.

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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

ACTION: With a measured RPF for a COMPARISON REGION less than 90% of the calculated RPF, either:

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

b. Increase the inferred INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE for the regions being controlled by a COMPARISON REGION, by increasing the inferred primary coolant temperature rise (INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE minus CORE AVERAGE INLET TEMPERATURE), 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 (RPFs) and the percent RPF discrepancies shall be determined according to the following schedule for each REFUELING CYCLE:

a. Calculated RPFs:

1. Prior to startup after each refueling,

2. At 20 plus or minus 5 Effective Full Power Days (EFPD) after each refueling,

3. At 40 plus or minus 5 EFPDs after each refueling, and

SPECIFICATION SR 4.2.3 (Continued)

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

b. Percent RPF Discrepancies, $\frac{(\text{RPF meas} - \text{RPF calc})}{(\text{RPF calc})} \times 100$:

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

2. Within a total elapsed time of 10 calendar days at reactor power levels above 40% RATED THERMAL POWER after completing any of the calculated RPF determinations 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 determination, the Percent RPF Discrepancy evaluation is not required, but the total elapsed time at reactor power levels above 40% of RATED THERMAL POWER between Percent RPF Discrepancy calculations shall not exceed 45 calendar days.

BASIS FOR SPECIFICATION LCO 3.2.3/SR 4.2.3

Use of COMPARISON REGIONS requires that conditions in the COMPARISON REGION (power, flow rate, and outlet temperature) be known. Region Peaking Factor (RPF) 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, RPF discrepancies up to 10% (positive or negative) are not unexpected or considered to be excessive. Using the COMPARISON REGION method of operation, only excessively negative RPF 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 be at least as large as 90% of the calculated RPF ensures that any region being used as a COMPARISON REGION will not have a large negative RPF discrepancy, i.e., less than minus 10%.

The percent RPF discrepancy is defined as follows:

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

The calculated RPFs 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 RPFs in Regions 20 and 32 through 37 to the calculated RPFs 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 RPFs at the specified frequency will ensure that appropriate RPFs continue to be used in determining the region outlet temperature for Regions 20 and 32 through 37.

In using the COMPARISON REGION method, it is assumed that the INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE for the region being controlled by a COMPARISON REGION is incorrect. The correct value is determined from a calculation involving power and flow rate and the primary coolant temperature rise in the COMPARISON REGION as follows:

> INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE equals the inferred region temperature rise plus the CORE AVERAGE INLET TEMPERATURE.

BASIS FOR SPECIFICATION LCO 3.2.3/SR 4.2.3 (Continued)

The inferred region temperature rise equals the ratio of calculated RPFs:

$$\frac{\text{RPF (REGION)}}{\text{RPF (COMPARISON REGION)}}$$

times the ratio of the primary coolant flow rates:

$$\frac{\text{Flow (COMPARISON REGION)}}{\text{Flow (REGION)}}$$

times the primary coolant temperature rise for the COMPARISON REGION.

Therefore, if the COMPARISON REGION has an excessively negative RPF discrepancy, the inferred region temperature rise will be too low and has to be increased to compensate. The ACTION statement ensures this compensation.

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

CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

3/4.2.4 CORE INLET ORIFICE VALVES/MINIMUM HELIUM FLOW AND MAXIMUM CORE REGION TEMPERATURE RISE

LIMITING CONDITION FOR OPERATION

3.2.4 The total helium circulator flow or the helium coolant temperature rise for all core regions shall be maintained within the limits given in Table 3.2.4-1.

APPLICABILITY: LOW POWER*, STARTUP, SHUTDOWN**, AND REFUELING**

ACTION:

a. In LOW POWER or STARTUP, with any of the above limits exceeded, either:

1. Correct the out-of-limit condition within 15 minutes, or

2. Be in at least SHUTDOWN within 1 hour with the inlet orifice valves adjusted for equal region coolant flows within the following 8 hours and, if the applicable limits are still exceeded, initiate PCRV depressurization within the following 1 hour.

b. In SHUTDOWN** or REFUELING** with the inlet orifice valves adjusted for equal region flows, with any of the above limits exceeded, either:

1. Correct the out-of-limit condition within 15 minutes, or

2. Initiate PCRV depressurization within the time limits of Figure 3.4.1-3 of Specification 3/4.4.1.

---

* Applicable for power levels for which limits are shown in Figures 3.2.4-1, -3, and -5.

** Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

SPECIFICATION LCO 3.2.4 (Continued)

c. In SHUTDOWN** or REFUELING** with the inlet orifice valves adjusted at any other position, with any of the above limits exceeded, either:

1. Adjust the inlet orifice valves to equal region coolant flows and be within the above limits within 8 hours, or

2. Initiate PCRV depressurization within the time limits of Figure 3.4.1-3 of Specification 3/4.4.1.

SURVEILLANCE REQUIREMENTS

4.2.4 At least once per 12 hours the total helium circulator flow or the helium coolant temperature rise through each core region shall be determined to be within the above limits.

** Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

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Table 3.2.4-1

Region Orifice Position	Reactor Pressure Helium Density	Limiting Condition for Operation
All regions set for equal region coolant flow*** EXCEPT Up to 10 regions may have their orifices further open.	Greater than 50 psia, with helium density* greater than 60%, but less than, or equal to, 107.5%.	The total helium circulator flow shall be greater than or equal to the minimum allowable value shown in Figure 3.2.4-1 or 3.2.4-2.
As above	Greater than 50 psia, with helium density* less than, or equal to, 60%.	The total helium circulator flow shall be greater than, or equal to, the minimum allowable value shown in Figures 3.2.4-3 or 3.2.4-4.
All regions set for equal region coolant flow***.	Less than or equal to 50 psia.	The helium coolant temperature rise** through any core region shall not exceed 600 degrees F.
Orifice valves at any position (Adjusted for nominal equal region outlet temperature).	Greater than 50 psia.	The helium coolant temperature rise** through any core region shall not exceed the limit shown in Figure 3.2.4-5.
Orifice valves at any position (Adjusted for nominal equal region outlet temperature).	Less than or equal to 50 psia.	The helium coolant temperature rise** through any core region shall not exceed 350 degrees F.

* Percent helium density equals:

$$\frac{175.12 \times \text{Reactor Pressure (psia)}}{\text{(Circulator inlet temperature (degrees F) plus 460)}}$$

** Helium coolant temperature rise equals INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE minus CORE AVERAGE INLET TEMPERATURE.

*** Equal region coolant flow with 7 column region orifice valves set between 8% and 20% open (or the corresponding position for 5 column regions).

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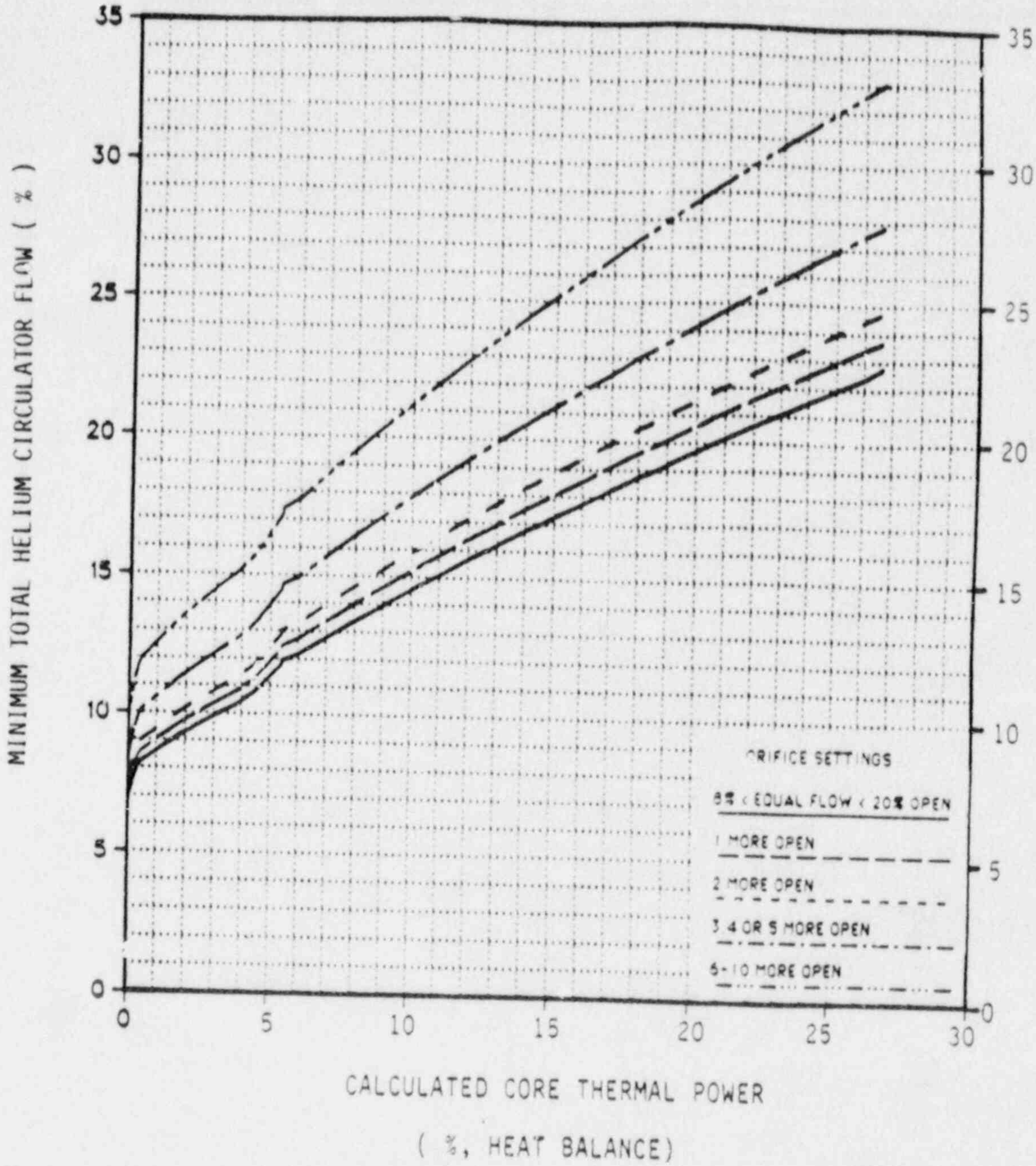
ORIFICE VALVES ADJUSTED FOR EQUAL REGION COOLANT FLOW

› 60% TO ≤ 107.5% HELIUM DENSITY

› 50 PSIA REACTOR PRESSURE



MINIMUM ALLOWABLE PRIMARY COOLANT FLOW

Figure 3.2.4-1

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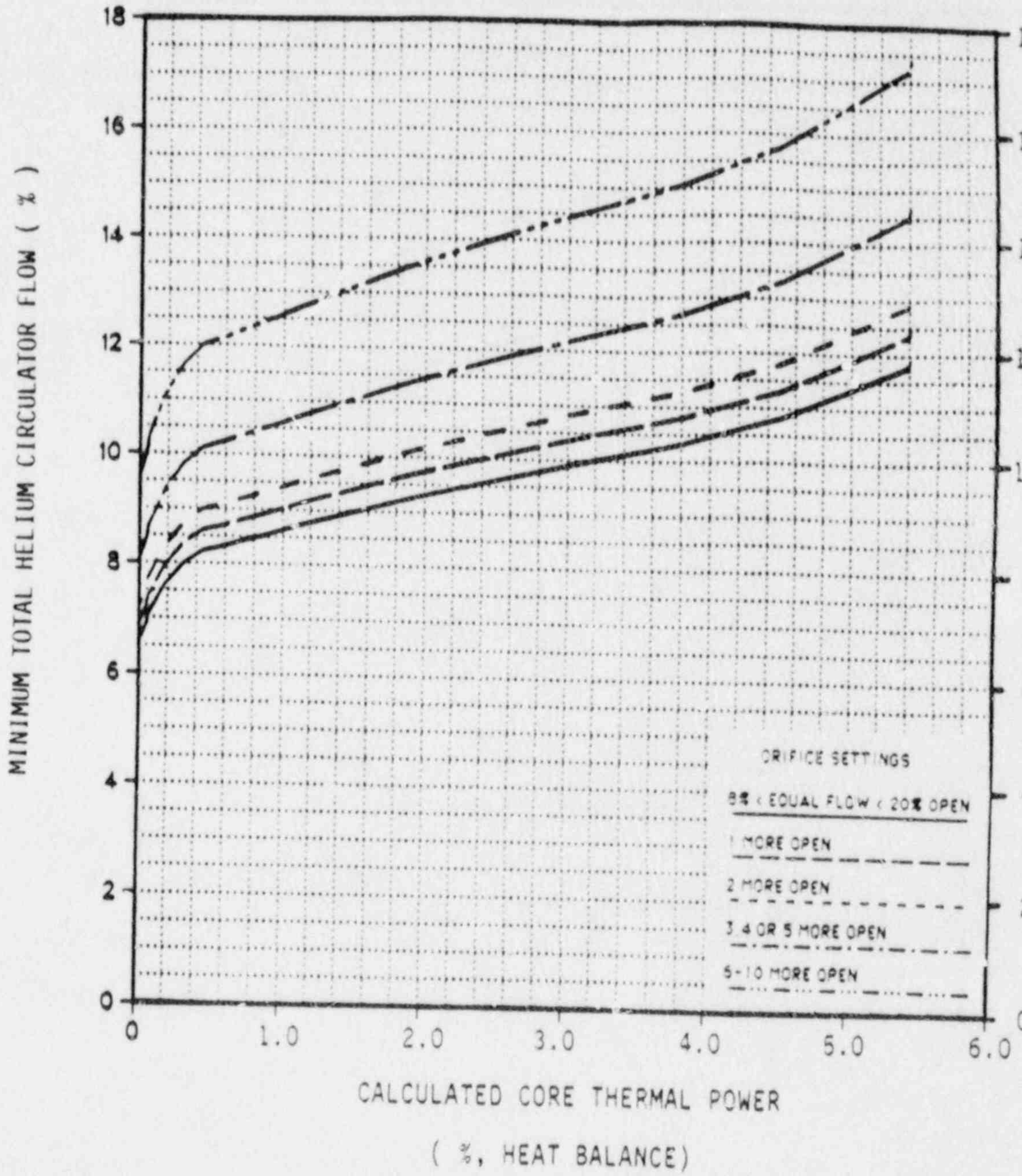
ORIFICE VALVES ADJUSTED FOR EQUAL REGION COOLANT FLOW

> 60% TO ≤ 107.5% HELIUM DENSITY

> 50 PSIA REACTOR PRESSURE



MINIMUM ALLOWABLE PRIMARY COOLANT FLOW
(EXPANDED RANGE)

Figure 3.2.4-2

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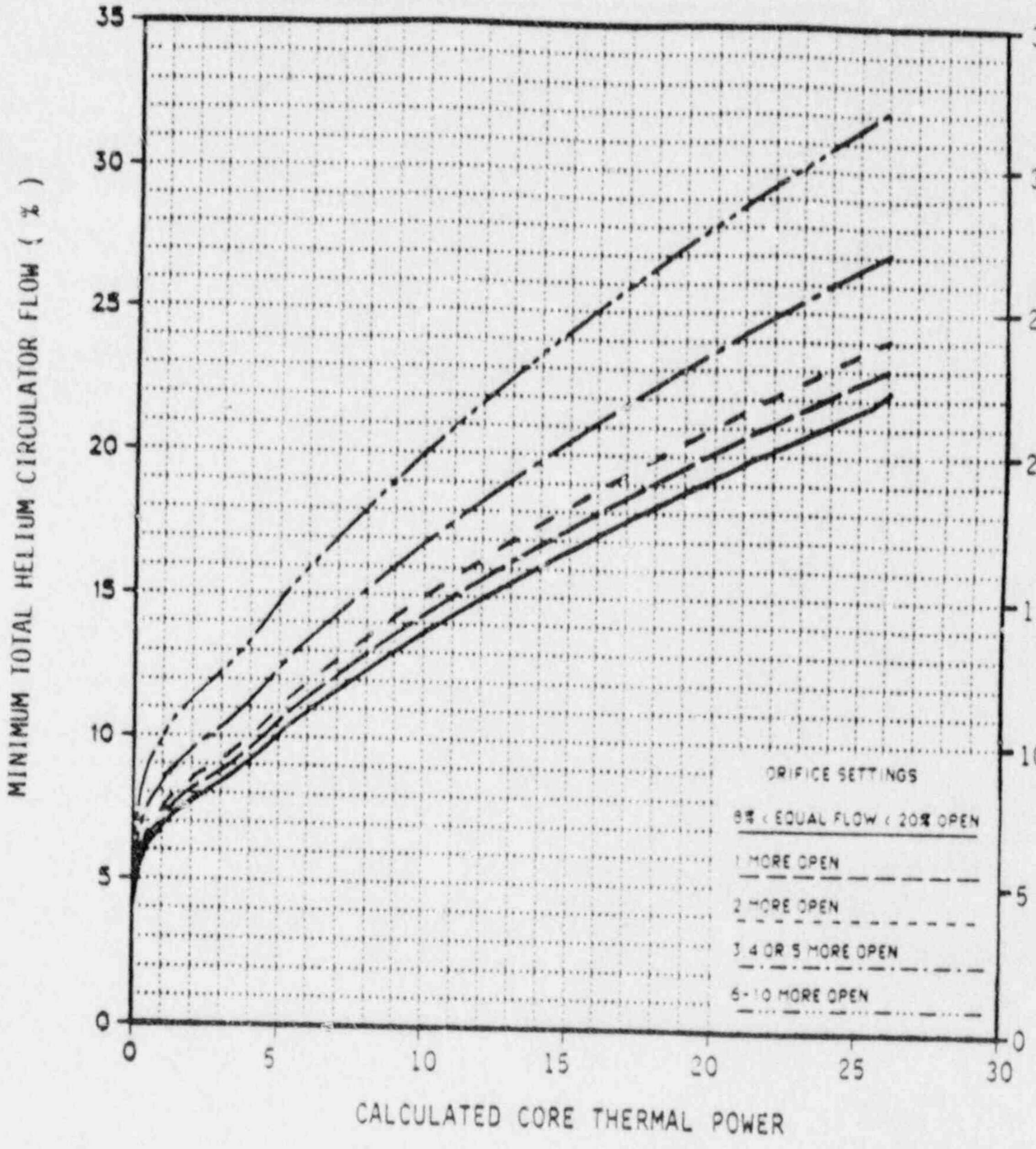
ORIFICE VALVES ADJUSTED FOR EQUAL REGION COOLANT FLOW

≤ 60% HELIUM DENSITY

> 50 PSIA REACTOR PRESSURE



MINIMUM ALLOWABLE PRIMARY COOLANT FLOW

Figure 3.2.4-3

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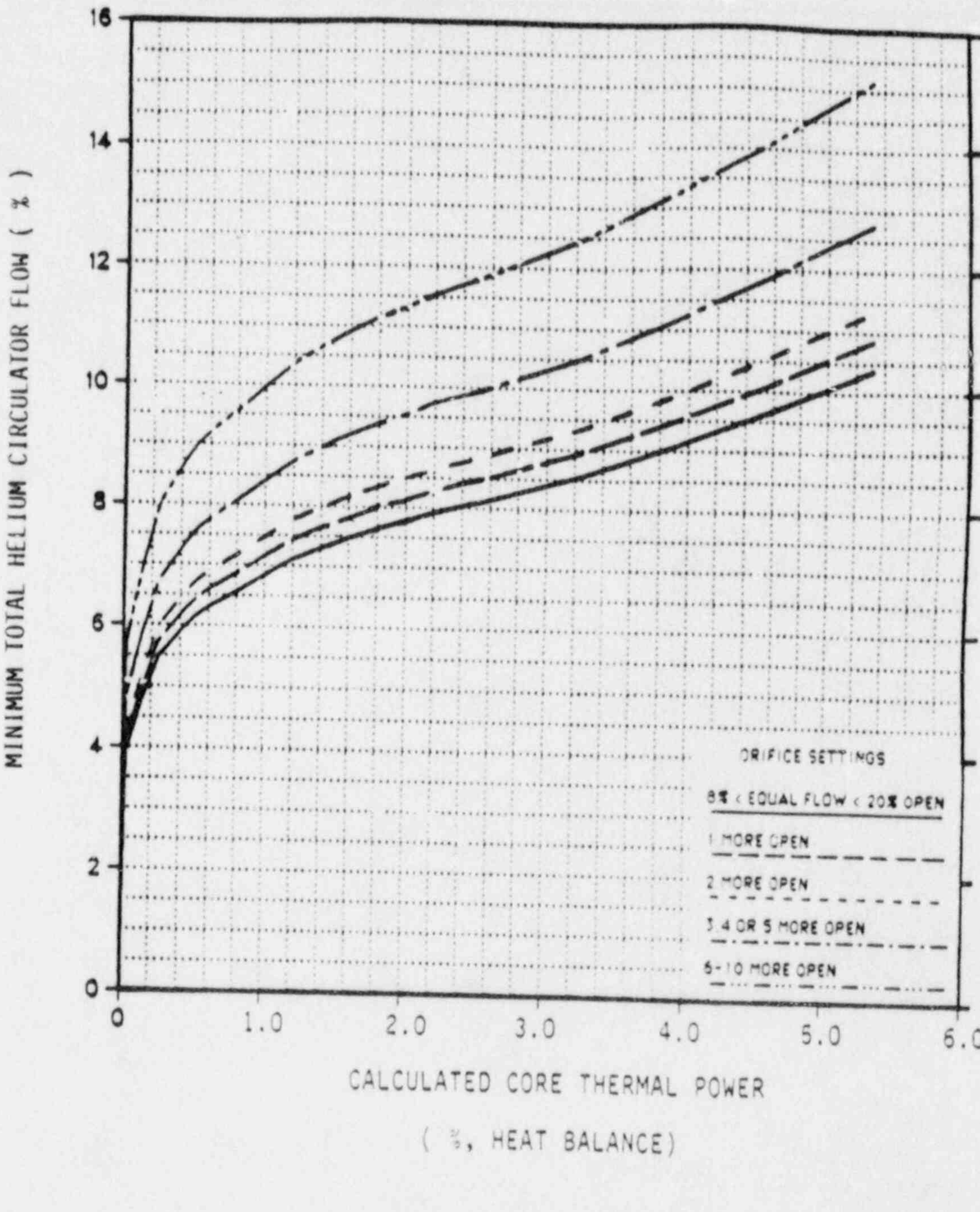
ORIFICE VALVES ADJUSTED FOR EQUAL REGION COOLANT FLOW
≤ 60% HELIUM DENSITY
> 50 PSIA REACTOR PRESSURE



MINIMUM ALLOWABLE PRIMARY COOLANT FLOW
(EXPANDED RANGE)

Figure 3.2.4-4

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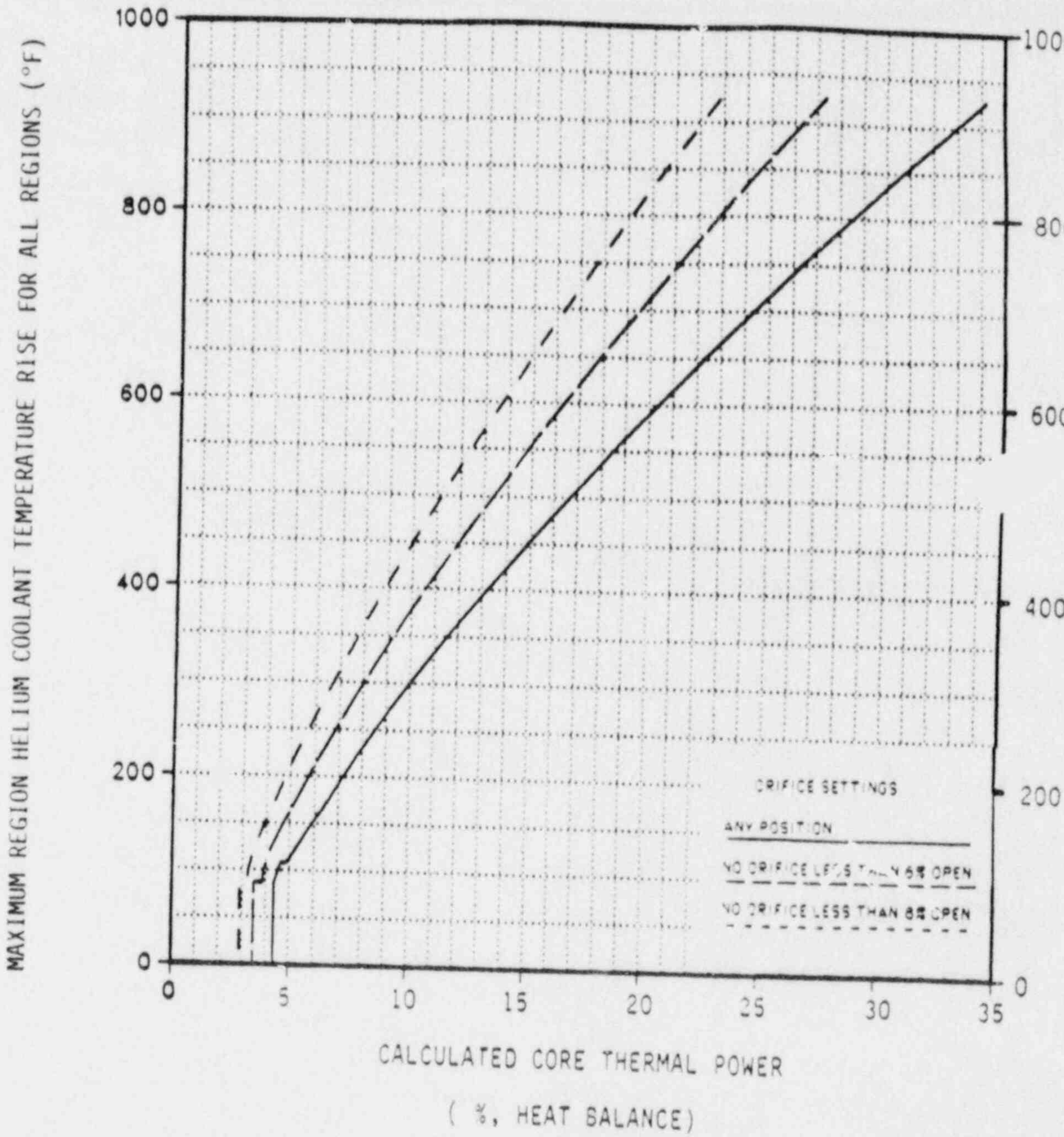
ORIFICE VALVES ADJUSTED FOR

NOMINAL EQUAL REGION OUTLET TEMPERATURES

≤ 107.5% HELIUM DENSITY

> 50 PSIA REACTOR PRESSURE



MAXIMUM ALLOWABLE REGION HELIUM COOLANT TEMPERATURE RISE

Figure 3.2.4-5

BASIS FOR SPECIFICATION LCO 3.2.4/SR 4.2.4

The minimum helium circulator 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 or reverse flow, which could result in excessive fuel temperatures.

This specification addresses minimum flow requirements for all coolant channels. Since low coolant flows exist at lower reactor powers, its applicability is limited to less than approximately 25% RATED THERMAL POWER. Specific power level end points for given conditions are as shown on the Figures. Since THERMAL POWER is continuously generated by decay heat even after the reactor is shut down, the flow requirements are also applicable in SHUTDOWN and REFUELING, whenever the CALCULATED BULK CORE TEMPERATURE exceeds 760 degrees F.

This specification is not applicable during SHUTDOWN or REFUELING when the CALCULATED BULK CORE TEMPERATURE is less than 760 degrees F. Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of primary coolant helium flow.

The applicability of this specification is also limited to the range of power level indicated in Figures 3.2.4-1, 3.2.4-3, and 3.2.4-5. Above the power levels for which limits are shown in these Figures, the Reactor Core Safety Limit, Specification 2.1.1, governs. In addition to this specification, fuel integrity is ensured for power levels from 0 to 100% by limiting the INDIVIDUAL REFUELING REGION OUTLET TEMPERATURES to values given in Specification 3.2.2. The core flow fraction limits shown in Figures 3.2.4-1, 3.2.4-2, 3.2.4-3, and 3.2.4-4 are based on, and thus valid for, equal region coolant flow orifice settings within the range of 8% to 20% open for seven column regions and the corresponding settings for five column regions, i.e., within the range of 4.4% to 13.4% open for five column regions. Equal region outlet temperature orificing is precluded below about 3% power by Figure 3.2.4-5 because uncertainties in instrumentation exceed the allowable temperature rise.

BASIS FOR SPECIFICATION LCO 3.2.4/SR 4.2.4 (Continued)

The limits of this specification have been developed based upon a number of conservative assumptions. For the limits in Figures 3.2.4-1, 3.2.4-2 and 3.2.4-5, it was assumed that the primary system was pressurized to 107.5 percent of design helium density. At lower densities, higher region temperature rises and lower primary coolant flow are acceptable. Since startup operations can proceed with lower helium densities, after the reactor has been pressurized to greater than 100 psia, which corresponds to about 30 percent helium inventory at 200 degrees F, flow requirements were calculated for 60% helium density and are given in Figures 3.2.4-3 and 3.2.4-4.

Percent helium density equals

$$\frac{175.12 \times \text{Reactor Pressure (psia)}}{\text{Circulator inlet temperature (degrees F) plus 460}}$$

The core inlet helium temperature used in the analysis covers the range of 100-400 degrees F between 5% and 25% RATED THERMAL POWER and 100-700 degrees F above 5% RATED THERMAL POWER. These are reasonable assumptions for low power operation.

The analysis is based on operation of two circulators between 0 and 5% RATED THERMAL POWER and four circulators above 5% RATED THERMAL POWER. This is consistent with plant operation.

In the analysis to determine the limits, the effects of heat conduction between columns in a region, or between regions, were conservatively neglected. Envelope values of RPF/Intra Region Peaking (3.0/1.25 and 1.6/1.61) were used to anticipate worst case conditions considering all future fuel cycles. Consistently conservative nominal values and uncertainties were used for bypass flows and measured parameters throughout the analysis. For the condition with orifice valves at any position, the allowable region delta T is based upon a region peaking factor equal to 0.4. For regions with higher peaking factors, higher region delta T's are acceptable.

The circulator flow determination is normally based on the empirical relationship between flow and circulator inlet nozzle delta P, local temperature, and local pressure. The uncertainties associated with control room indication of these parameters were accounted for in the analysis. Other flow determination methods are acceptable provided the associated uncertainties are accounted for and the calculated circulator flow is adjusted accordingly.

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BASIS FOR SPECIFICATION LCO 3.2.4/SR 4.2.4 (Continued)

Besides the minimum flow requirement curves with the orifices set for equal region flows in Figures 3.2.4-1, 3.2.4-2, 3.2.4-3, and 3.2.4-4, flow requirements are provided with up to 10 orifice valves positioned further open. These curves assist in the transition between equal region flows and equal region outlet gas temperatures. By monitoring the total circulator flow when the orifices are adjusted for equal region coolant flows, minimum flow through each region at the appropriate power can be ensured. When the orifice valves are adjusted to different positions, minimum coolant flows can be ensured for each region by monitoring the helium coolant temperature rise in that region. Maximum temperature rise requirement curves are presented for the case where the inlet orifice valves are adjusted to any position as well as the cases where no seven-column orifice valve is closed to less than 8% or less than 6% open (or the corresponding position for five-column regions).

For depressurized operations, helium coolant temperature rise 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.

To ensure that flow stagnation in a fuel column or region does not persist, an ACTION time of only 15 minutes is allowed to correct the out of limit condition.

The requirement to be in SHUTDOWN within 1 hour with the orifices set for equal flows in an additional 8 hours is realistic because it takes approximately 4 - 6 hours to set the orifice valves from equal temperatures to equal region flows. This is considered acceptable since there is sufficient primary coolant flow from the circulators which are driven by steam generated from residual heat in the system following reactor shutdown until the auxiliary boilers can be brought on line. If, after this action, there is still inadequate flow, depressurizing the PCRV further reduces the tendancy toward stagnation and reverse flow. Primary coolant flow is maintained to the extent possible and depressurization is continued until applicable limits can be met.

Surveillance of the helium circulator flow or helium coolant temperature rise once per 12 hours ensures that the LCO requirements are met. In addition, plant procedures require that the flow rate, core outlet temperatures, and power level be monitored continuously whenever the power level is being changed or orifice valves are being adjusted. In performance of the surveillance, the total helium circulator flow is determined by calculation consistent with the method used to determine the required flow for the analysis.

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. All RCDs in place on top of the core,

b. All RCD pins engaged in the plenum elements, and

c. RCD structural integrity maintained.

APPLICABILITY: POWER, LOW POWER, and STARTUP

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

SURVEILLANCE REQUIREMENTS

4.2.5 The Region Constraint Devices (RCDs) shall be demonstrated OPERABLE during each refueling outage by:

a. Visually inspecting the upper core plenum from those regions being refueled to ensure that the RCDs within visible range are in place on top of the core.

b. Monitoring the fuel handling machine vertical location coordinates and lifting force as RCDs are being removed and installed, to ensure that the RCD pins engage in the plenum elements and that they disengage as expected. Upon reinstallation, the vertical location of the fuel handling machine shall be within 3 inches of the expected coordinates to verify proper engagement.

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

BASIS FOR SPECIFICATION LCO 3.2.5/SR 4.2.5

Region Constraint Devices (RCDs), located on top of plenum elements 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 (FSAR Section 3.3.1.1). The RCDs are used to limit horizontal 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 performing their design function by restraining the fuel columns.

Monitoring the lifting force required 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.

Monitoring the vertical location coordinate ensures the RCDs have been properly installed with the pins engaged in the plenum elements.

CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

3/4.2.6 POWER-TO-FLOW RATIO

LIMITING CONDITION FOR OPERATION

3.2.6 The POWER-TO-FLOW RATIO shall be maintained below the curve of Figure 3.2.6-1.

APPLICABILITY: POWER and LOW POWER*

ACTION: During any individual transient, if the maximum POWER-TO-FLOW RATIO exceeds the curve of Figure 3.2.6-1:

a. With the maximum POWER-TO-FLOW RATIO less than or equal to 1.17:

   1. Restore the plant to below the curve of Figure 3.2.6-1 within 30 minutes, or

   2. Be in at least STARTUP within the next 12 hours, and

   3. Perform the evaluation required by ACTION c below.

b. With the maximum POWER-TO-FLOW RATIO greater than 1.17:

   1. Reduce power so that the POWER-TO-FLOW RATIO is reduced below 1.17 within 2 minutes plus any delay time defined in ACTION c.1 below, or

   2. Be in SHUTDOWN within the next 5 minutes, and

   3. Perform the evaluation required by ACTION c below.

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* Applicable only above 15% RATED THERMAL POWER

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SPECIFICATION LCO 3.2.6 (Continued)

c. Determine the P/F Integral Fraction of Allowable Operating Time.

As soon as practicable, but no more than 12 hours after any individual transient with the maximum POWER-TO-FLOW RATIO (P/F RATIO) exceeding Figure 3.2.6-1, for each fuel segment within the core:

1. Determine the fraction of the Allowable Operating Time specified in Figure 3.2.6-2, for each POWER-TO-FLOW RATIO (or interval of POWER-TO-FLOW RATIOS, to simplify the calculations) experienced during the transient, as follows:

a) POWER-TO-FLOW RATIOS Less Than or Equal to 2.5

The fraction of Allowable Operating Time for each P/F RATIO interval experienced during each transient shall be the time period that the POWER-TO-FLOW RATIO exceeds the limit of Figure 3.2.6-1, divided by the Allowable Operating Time for that P/F RATIO interval (per Figure 3.2.6-2).

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

The fraction of Allowable Operating Time for this P/F RATIO interval experienced during each transient shall be that time period from the point where the POWER-TO-FLOW RATIO exceeds the limit of Figure 3.2.6-1, until it drops below 2.5, not including the first 100 seconds, divided by the Allowable Operating Time for this P/F RATIO interval (per Figure 3.2.6-2). The calculation of additional fractions for POWER-TO-FLOW RATIOS less than 2.5 are given in c.1.a above.

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c) POWER-TO-FLOW RATIOS Greater Than 15:

The fraction of Allowable Operating Time for this P/F RATIO interval experienced during each transient shall be that time period from the point where the POWER-TO-FLOW RATIO exceeds the limit of Figure 3.2.6-1, until it drops below 2.5, not including the first 60 seconds, divided by the Allowable Operating Time for this P/F RATIO interval (per Figure 3.2.6-2). The calculation of additional fractions for POWER-TO-F.¯W RATIOS less than 2.5 are given in c.1.a above.

2. Determine the P/F Integral Fraction of Allowable Operating Time by summing the fractions of Allowable Operating Time for each P/F RATIO interval determined above, accumulated over the lifetime of each fuel segment within the core.

3. Verify that the P/F Integral Fraction of Allowable Operating Time is less than or equal to 1.0, consistent with the Reactor Core SAFETY LIMIT of Specification 2.1.1.

SURVEILLANCE REQUIREMENTS

4.2.6 a The POWER-TO-FLOW RATIO shall be determined to be below the curve of Figure 3.2.6-1 at least once per 12 hours.

b. Within 12 hours after any operating transient where the POWER-TO-FLOW RATIO exceeds the limit of Figure 3.2.6-1, determine the P/F Integral Fraction of Allowable Operating Time per Specification 3.2.6, ACTION c.

c. At least once per 7 days, determine the P/F Integral Fraction of Allowable Operating Time per Specification 3.2.6, ACTION c.

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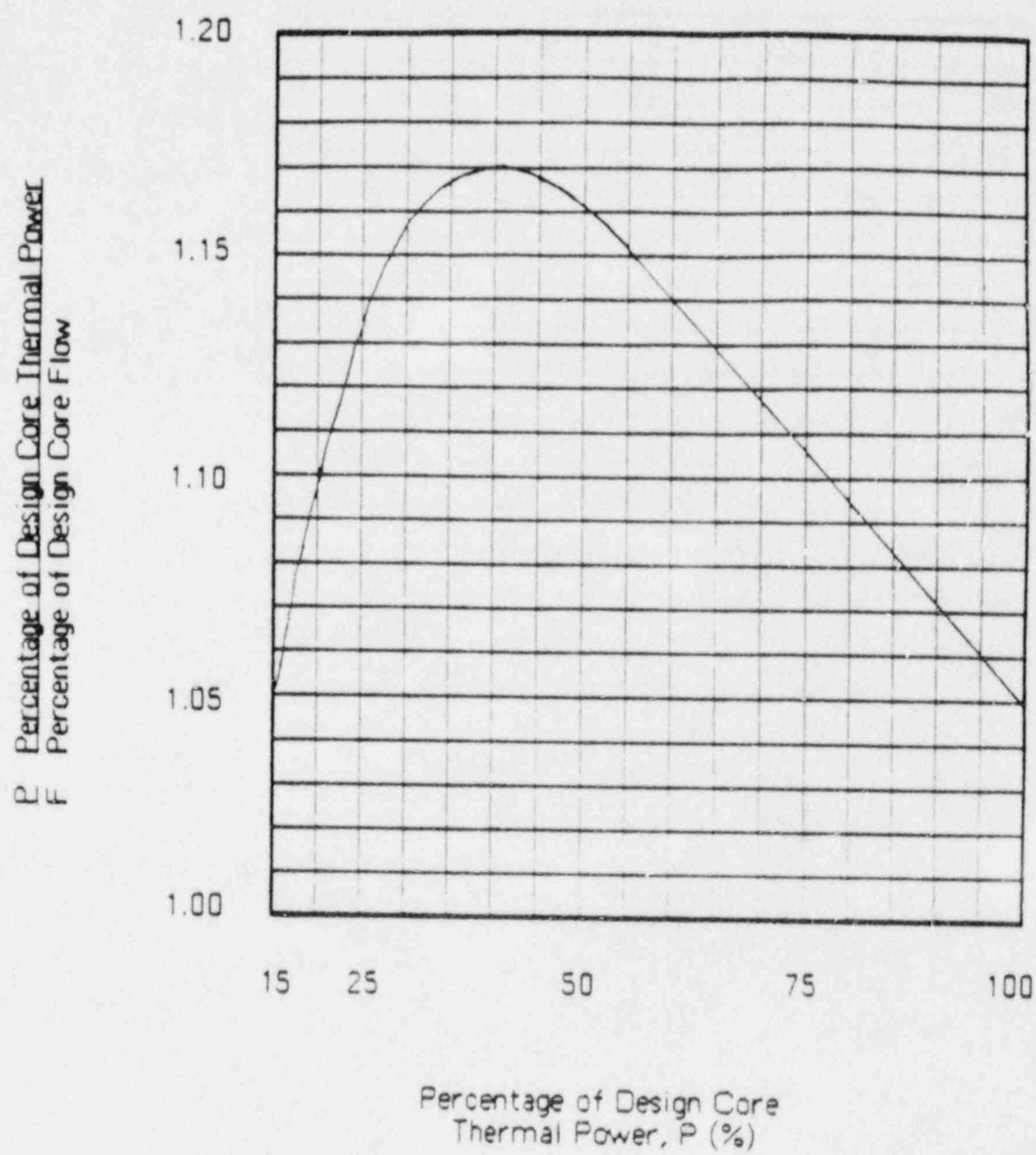

POWER - TO - FLOW RATIO LIMIT

Figure 3.2.6-1

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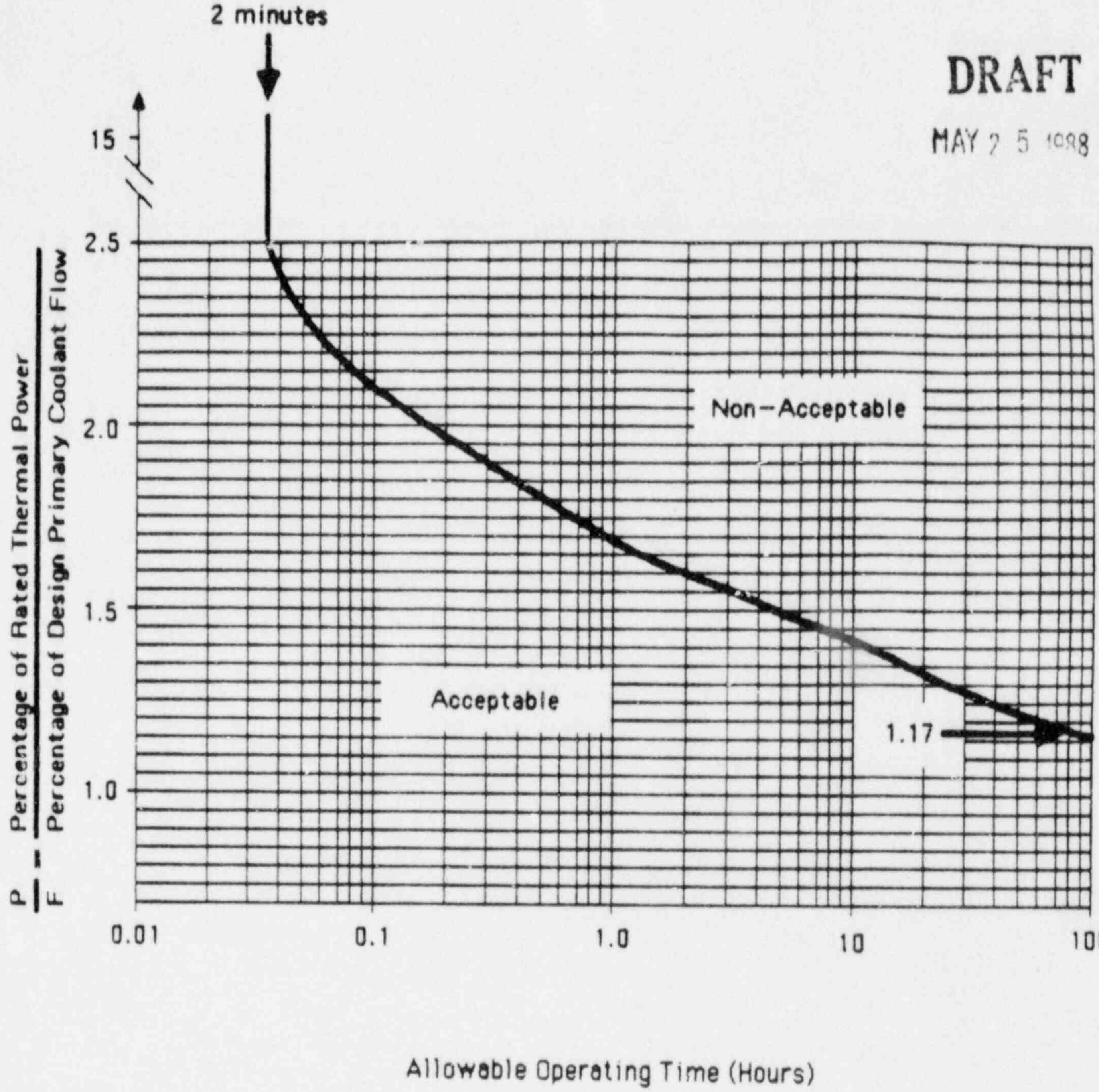

ALLOWABLE OPERATING TIMES WITH
POWER - TO - FLOW RATIOS EXCEEDING FIGURE 3.2.6-1

Figure 3.2.6-2

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BASIS FOR SPECIFICATION LCO 3.2.6/SR 4.2.6

SL 2.1.1 limits the P/F Integral Fraction of Allowable Operating Time of the summation of a number of individual transients. The individual transients are limited by Specification 3.2.6. Further discussion on POWER-TO-FLOW RATIO limits is provided in FSAR section 3.6.8 and in the Basis for SL 2.1.1.

Calculational Methods and Assumptions for Curves of Figures 3.2.6-1 and 3.2.6-2.

In Figures 3.2.6-1 and 3.2.6-2, the quantity P (percent 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 rate) is the total coolant flow rate measured at the circulators (in lb/hr) divided by 3.5E+06 lb/hr, and multiplied by 100%.

The limiting combinations of percentage of RATED THERMAL POWER and percentage of design PRIMARY 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 region power peaking factor (RPF) is defined to be equal to the ratio of the region average power density (Preg) divided by the core average power density (P). Conservative values for the calculated RPFs were used consistent with the allowable limits specified in DESIGN FEATURE 5.3.4.c for a CORE AVERAGE OUTLET TEMPERATURE above 1250 degrees F. 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 rod pairs 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 rod pairs fully inserted or withdrawn, and 1.23 plus or minus 0.12 for regions with control rod pairs inserted more than 2 feet.

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BASIS FOR SPECIFICATION LCO 3.2.6/SR 4.2.6 (Continued)

The measured INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE for the nine regions with their orifice valves most fully closed and all regions with control rod pairs inserted more than 2 feet, was assumed to be not more than 50 degrees F greater than the CORE AVERAGE OUTLET TEMPERATURE, consistent with Specification 3.2.2.a.1.a. The measured INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE for the remaining core regions was conservatively assumed to be up to 200 degrees F greater than the CORE AVERAGE OUTLET TEMPERATURE, (specification 3.2.2.a.2 and Figure 3.2.2-1). A measurement uncertainty for the core region outlet temperature of plus or minus 50 degrees F was assumed, and 5% uncertainty in flow rate measurement and a 5% uncertainty in reactor THERMAL POWER measurement were assumed in establishing the limit consistent with FSAR Section 3.6.7. The 95% confidence interval on experimental data was used in the most conservative manner to determine the rate of migration of the fuel kernel as a function of the fuel kernel temperature and the average temperature gradient across the fuel kernel.

For the total fuel lifetime in the core, based on calculation incorporating plant parameters and uncertainties appropriate for longer time, 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 3.2.6-1. Out of a total inner coating thickness of 70 microns, only 50 microns have been used for the determination of fuel particle failure in establishing the limit curve in Figure 3.2.6-2.

Determination of Integral Fraction in ACTION c.2

The Integral Fraction of Allowable Operating Times is determined as follows.

1. The range of possible POWER-TO-FLOW RATIOS above the limit of Figure 3.2.6-1 is divided into intervals, for ease of calculation.

2. The Allowable Operating Time above the limit of Figure 3.2.6-1 is determined each P/F RATIO interval from Figure 3.2.6-2.

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BASIS FOR SPECIFICATION LCO 3.2.6/SR 4.2.6 (Continued)

3. For each transient, the actual time period during which the P/F RATIO exceeded the limit of Figure 3.2.6-1 for each interval is divided by the Allowable Operating Time for that interval.

4. The individual fractions determined in Step 3 above are summed for each fuel segment, over its lifetime in the core. This is the Integral Fraction of Allowable Operating Time which may not exceed 1.0, per SL 2.1.1.

BASIS for POWER-TO-FLOW RATIOS Less Than or Equal to 1.17

For an individual transient with a maximum POWER-TO-FLOW RATIO above the curve of Figure 3.2.6-1 and less than or equal to 1.17, a 30 minute limit has been established from an operating viewpoint as adequate for reactor operator action. This provides sufficient conservatism since Figure 3.2.6-2 allows a total of 100 hours for the Integrated Operating Time of all such transients. If the transient is not reduced below Figure 3.2.6-1 within 30 minutes, an orderly reduction in power to at least STARTUP is appropriate.

BASIS for POWER-TO-FLOW RATIOS Greater Than 1.17 and Less Than or Equal to 2.5

The minimum time to prevent exceeding the curve of Figure 3.2.6-2 is 2 minutes, which occurs at POWER-TO-FLOW RATIOS of 2.5.

To reach a POWER-TO-FLOW RATIO of this magnitude through an increase in core power, significant equipment malfunction or failure, and/or significant deviations from operating procedures would have to occur.

Therefore, a 2 minute limit on individual transients is sufficiently conservative. For example, as can be seen from Figure 3.2.6-2, sufficient time (at least 9 minutes) is available for the reactor operator to take corrective action to prevent the core SAFETY LIMIT from being exceeded for POWER-TO-F[illegible] RATIOS less than or equal to 2.0.

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BASIS FOR SPECIFICATION LCO 3.2.6/SR 4.2.6 (Continued)

BASIS for POWER-TO-FLOW RATIOS Above 2.5

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 temperature coefficient of reactivity provides an instrinsic means to reduce the core power and the POWER-TO-FLOW RATIO, and the PLANT PROTECTIVE 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.

Under fast transient conditions, either abnormal rapid power increases or sudden flow rate decreases, the Allowable Operating Times of Figure 3.2.6-2, which were derived from steady state calculations, are not a meaningful indicator of fuel kernel migration and fuel particle integrity, i.e., they are overly conservative. Accordingly, a delay period is appropriate for such transients. 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 curve of Figure 3.2.6-2. This delay period can be allowed without compromising the integrity of the fuel. 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 analyses results shown in Chapter 14 and Appendix D. As a result of many transient analyses, including those shown in FSAR Section 14 and Appendix D, the delay period has been conservatively set at 100 seconds for transients resulting in a maximum POWER-TO-FLOW RATIO above 2.5 but less than or equal to 15, and 60 seconds if the maximum POWER-TO-FLOW RATIO is greater than 15. See FSAR section 3.6.8.

BASIS for APPLICABILITY

APPLICABILITY is limited to power levels above 15% RATED THERMAL POWER, in that Figure 3.2.6-1 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.

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BASIS FOR SPECIFICATION LCO 3.2.6/SR 4.2.6 (Continued)

BASIS for 12 Hours for Integral Evaluation in ACTION c.

Following a deviation from the normal power-to-flow operating range, up to 12 hours are allowed by ACTION statement c to perform an evaluation, determine if the P/F Integral Fraction of Allowable Operating Time limit 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 the P/F Integral Fraction caused by a transient has caused a conservatively determined limit to be exceeded, because the plant will have either been shut down or restored to normal conditions. An additional 12 hours of operation under normal conditions will not significantly contribute to any additional fuel failures.

3/4.3 INSTRUMENTATION

3/4.3.1 PLANT PROTECTIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the instrumentation identified in Part 2 of Tables 3.3.1-1 through 3.3.1-4 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 3.3.1-4.

APPLICABILITY: At all times, unless permitted to be bypassed per Part 2 of Tables 3.3.1-1 through 3.3.1-4.

ACTION:

a. With a TRIP SETPOINT "as found" value less conservative than the value shown in the ALLOWABLE VALUE column of part. 1 of Tables 3.3.1-1 through 3.3.1-4, apply the applicable ACTION statement requirement of ACTION b below and comply with the requirements for REPORTABLE EVENTS.

b. If the minimum OPERABLE channels or the minimum degree of redundancy for each functional unit identified in part 2 of a table cannot be met or cannot be bypassed under the stated permissible bypass conditions, the following ACTION shall be taken:

1. For Table 3.3.1-1, the reactor shall be SHUTDOWN within 24 hours, except that to facilitate maintenance on the PLANT PROTECTIVE SYSTEM (PPS) moisture monitors, 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. During the time in which the TRIP functions are disabled the requirements of Specifications 3.4.3 and 3.4.4 shall be met and primary coolant shall not exceed a moisture concentration of 100 ppmv.

SPECIFICATION 3.3.3 (Continued)

2. For Table 3.3.1-2, the affected loop shall be shut down within 12 hours.

3. For Table 3.3.1-3, except for Functional Units 10a, 10b, 10c, and 10d, perform one of the following within 12 hours:

   a) The reactor shall be SHUTDOWN, or

   b) The affected helium circulator shall be shut down.

4. For Table 3.3.1-3, for Functional Units 10a, 10b, 10c, and 10d, apply the applicable ACTION statement requirements identified in that Table.

5. For Table 3.3.1-4, the reactor shall be SHUTDOWN within 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.1 Each channel, interlock, and automatic TRIP logic shall be demonstrated OPERABLE by the performance of the instrumentation surveillance and CHANNEL CALIBRATION requirements specified in Tables 4.3.1-1 through 4.3.1-4 and associated notes.

TABLE 3.3.1-1 (Part 1)

INSTRUMENT OPERATING REQUIREMENTS 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
2.	Startup Channel-High Count Rate	≤ 8.3E+04 cps	≤ 9.3E+04 cps
3a.	Linear Channel-High Channels 3,4,5 (Neutron Flux)	---------See Table 2.2.1-1-------	
3b.	Linear Channel-High Channels 6,7,8 (Neutron Flux)	---------See Table 2.2.1-1-------	
4.	Primary Coolant Moisture -High Level Monitor	≤ 60.5 degrees F dewpoint	≤ 62.2 degrees F dewpoint
	-Loop Monitor	≤ 20.4 degrees F dewpoint	≤ 22.1 degrees F dewpoint
5.	Reheat Steam Temperature -High	≤ 1055 degrees F	≤ 1067 degrees F
6.	Primary Coolant Pressure -Programmed Low	---------See Table 2.2.1-2-------	
7.	Primary Coolant Pressure -Programmed High	---------See Table 2.2.1-2-------	
8.	Hot Reheat Header Pressure -Low	≥ 44 psig	≥ 43 psig
9.	Main Steam Pressure-Low	≥ 1529 psig	≥1517 psig
10.	Plant Electrical System-Loss	≥ 278V ≤ 31.5 seconds	≥ 266V ≤ 35 seconds
11.	Two Loop Trouble	Not Applicable	Not Applicable
12.	High Reactor Building Temperature (Pipe Cavity)	≤ 161 degrees F	≤ 166 degrees F

TABLE 3.3.1-1 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, SCRAM

NO.	FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS
1a.	Manual (Control Room)	1	0	None
1b.	Manual (Outside Ctrl Rm)	2 (d)	1	None
2.	Startup Channel-High Count Rate	2	1	Reactor Mode Switch in "RUN"
3a.	Linear Channel-High, Channels 3, 4, 5 (Neutron Flux)	2 (d)	1	None
3b.	Linear Channel-High, Channels 6, 7, 8 (Neutron Flux)	2 (d)	1	None
4.	Primary Coolant Moisture			
	-High Level Monitor	1 (d,1)	1(b)	None
	-Loop Monitor	2/Loop (d,1)	1/Loop	(f)
5.	Reheat Steam Temperature - High	2 (a,d)	1	None
6.	Primary Coolant Pressure - Programmed Low	2 (d,g)	1	Less Than 30% RATED THERMAL POWER
7.	Primary Coolant Pressure - Programmed High	2 (d,g)	1	None
8.	Hot Reheat Header Pressure - Low	2 (d)	1	Less Than 30% RATED THERMAL POWER
9.	Main Steam Pressure - Low	2 (d)	1	Less Than 30% RATED THERMAL POWER
10.	Plant Electrical System - Loss	2 (c,d)	1	None
11.	Two Loop Trouble	2	1	Reactor Mode Switch in "Fuel Loading"
12.	High Reactor Building Temperature (Pipe Cavity)	2 (d)	1	None

Notes are on Pages 3/4 3-14 and 3/4 3-15

TABLE 3.3.1-2 (Part 1)

INSTRUMENT OPERATING REQUIREMENTS
FOR THE PLANT PROTECTIVE SYSTEM, LOOP SHUTDOWN

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1a.	Loop 1 Shutdown Logic	Not Applicable	Not Applicable
1b.	Loop 2 Shutdown Logic	Not Applicable	Not Applicable
2a.	Circulator 1A and 1B Shutdown - Loop Shutdown Logic	Not Applicable	Not Applicable
2b.	Circulator 1C and 1D Shutdown - Loop Shutdown Logic	Not Applicable	Not Applicable
3a.	Steam Generator Penetration Overpressure, Loop 1	≤ 796 psig	≤ 801 psig
3b.	Steam Generator Penetration Overpressure, Loop 2	≤ 796 psig	≤ 801 psig
4a.	High Reheat Header Activity, Loop 1	< 3.2 mR/hr Above Background	< 3.5 mR/hr Above Background
4b.	High Reheat Header Activity, Loop 2	< 3.2 mR/hr Above Background	< 3.5 mR/hr Above Background
5a.	Low Superheat Header Temperature, Loop 1 (j)	≥ 798 degrees F	≥ 794 degrees F
5b.	Low Superheat Header Temperature, Loop 2 (j)	≥ 798 degrees F	≥ 794 degrees F
5c.	High Differential Temperature Between Loop 1 and Loop 2 (j)	≤ 44.8 degrees F	≤ 46.7 degrees F

Notes are on Pages 3/4 3-14 and 3/4 3-15

TABLE 3.3.1-2 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, LOOP SHUTDOWN

NO.	FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS
1a.	Loop 1 Shutdown Logic	2	1	None
1b.	Loop 2 Shutdown Logic	2	1	None
2a.	Circulator 1A and 1B Shutdown - Loop Shutdown Logic	2	1	None
2b.	Circulator 1C and 1D Shutdown - Loop Shutdown Logic	2	1	None
3a.	Steam Generator Penetration Overpressure, Loop 1	2 (d)	1	None
3b.	Steam Generator Penetration Overpressure, Loop 2	2 (d)	1	None
4a.	High Reheat Header Activity, Loop 1	2 (d)	1	None
4b.	High Reheat Header Activity, Loop 2	2 (d)	1	None
5a.	Low Superheat Header Temperature, Loop 1 (j)	2 (d)	1	Less Than 30% RATED THERMAL POWER
5b.	Low Superheat Header Temperature, Loop 2 (j)	2 (d)	1	Less Than 30% RATED THERMAL POWER
5c.	High Differential Temperature Between Loop 1 and Loop 2 (j)	2 (d)	1	Less Than 30% RATED THERMAL POWER

Notes are on Pages 3/4 3-14 and 3/4 3-15

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TABLE 3.3.1-3 (Part 1)

INSTRUMENT OPERATING REQUIREMENTS FOR THE PLANT PROTECTIVE SYSTEM,
CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1.	Circulator Speed - Low	$\leq$ 1830 rpm Below Normal As Programmed by Feedwater Flow	$\leq$ 2035 rpm Below Normal As Programmed by Feedwater Flow
2a.	Loop 1, Fixed Feedwater Flow - Low (Both Circulators)	$\geq$ 230,500 lb/hr (20% of normal Full Load)	$\geq$ 230,500 lb/hr (20% of normal Full Load)
2b.	Loop 2, Fixed Feedwater Flow - Low (Both Circulators)	$\geq$ 230,500 lb/hr (20% of normal Full Load)	$\geq$ 230,500 lb/hr (20% of normal Full Load)
3.	Loss of Circulator Bearing Water	$\geq$ 459 psid	$\geq$ 454 psid
4.	Circulator Penetration Trouble	$\leq$ 796 psig	$\leq$ 801 psig
5.	Circulator Drain Malfunction	$\geq$ 8.5 psid	$\geq$ 8.0 psid
6.	Circulator Speed - High Steam	$\leq$ 11,495 rpm	$\leq$ 11,684 rpm
7.	Manual	Not Applicable	Not Applicable
8.	Circulator Seal Malfunction	$\geq$ -5.2" w.g., $\leq$ +74.8" w.g.	$\geq$ -6.1" w.g., $\leq$ +76.1" w.g.
9.	Circulator Speed - High Water	$\leq$ 8,589 rpm	$\leq$ 8,786 rpm
10a.	Steam Leak Detection Turbine Building Loop 1	$\leq$ 52.3 degrees F per minute rate of rise	$\leq$ 52.8 degrees F per minute rate of rise
10b.	Steam Leak Detection Reactor Building Loop 1	$\leq$ 52.3 degrees F per minute rate of rise	$\leq$ 52.8 degres F per minute rate of rise

TABLE 3.3.1-3 (Part 1)

_ANT PROTECTIVE SYSTEM, TRIP SETPOINTS - CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
10c.	Steam Leak Detection Turbine Building Loop 2	$\leq$ 52.3 degrees F per minute rate of rise	$\leq$ 52.8 degrees F per minute rate of rise
10d.	Steam Leak Detection Reactor Building Loop 2	$\leq$ 52.3 degrees F per minute rate of rise	$\leq$ 52.8 degrees F per minute rate of rise

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TABLE 3.3.1-3 (Part 2)

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PROTECTIVE SYSTEM, CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS
1.	Circulator Speed - Low (k)	2 (d)	1	Less Than 30% RATED THERMAL POWER
2a.	Loop 1, Fixed Feed-water Flow - Low (Both Circulators)	2 (d)	1	Less Than 30% RATED THERMAL POWER
2b.	Loop 2, Fixed Feed-water Flow - Low (Both Circulators)	2 (d)	1	Less Than 30% RATED THERMAL POWER
3.	Loss of Circulator Bearing Water (k)	2 (d)	1	None
4.	Circulator Penetration Trouble (k)	2 (d)	1	None
5.	Circulator Drain Malfunction (k)	2 (d)	1	None
6.	Circulator Speed - High Steam (k)	2 (d)	1	None
7.	Manual	1	0	None
8.	Circulator Seal Malfunction (k)	2 (d)	1	Opposite loop shutdown <u>or</u> circulator seal malfunction trip of other circulator in same loop
9.	Circulator Speed - High Water	2 (d)	1	None

Notes are on Pages 3/4 3-14 and 3/4 3-15

SPECIFICATION LCO 3.3.1

TABLE 3.3.1-3 (Part 2)

PLANT PROTECTIVE SYSTEM, OPERABILITY REQUIREMENTS - CIRCULATOR TRIP

NO.	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	APPLICABILITY MODE	ACTION
10a.	Steam Leak Detection Turbine Building Loop 1	4	2	3*	Note 1	1,2,3
10b.	Steam Leak Detection Reactor Building Loop 1	4	2	3*	Note 1	1,2,3
10c.	Steam Leak Detection Turbine Building Loop 2	4	2	3*	Note 1	1,2,3
10d.	Steam Leak Detection Reactor Building Loop 2	4	2	3*	Note 1	1,2,3

Note:

1. The reactor shall not be operated in STARTUP (above 2% RATED THERMAL POWER), LOW POWER, or POWER except as provided by these requirements and their associated ACTION statements.

* 7 channels per building must be OPERABLE.

TABLE 3.3.1-3 (Part 2)

ACTION STATEMENTS (for 10a, 10b, 10c and 10d Only)

ACTION 1 - With only 7 OPERABLE channels in either building or in both buildings, place the inoperable channel in bypass within 1 hour or reduce power to below 2% RATED THERMAL POWER within the next 12 hours. The channel shall be returned to OPERABLE status within the following 7 days.

ACTION 2 - With only 6 OPERABLE channels in either building or in both buildings, place the inoperable channels in bypass and reduce power to below 2% RATED THERMAL POWER within the next 12 hours. Operation at power may continue if at least 7 OPERABLE channels in both buildings are placed in service.

ACTION 3 - With inoperable channels or loops other than as provided in ACTION 1 and 2 above, reduce power to below 2% RATED THERMAL POWER within the next 12 hours.

TABLE 3.3 1-4 (Part 1)

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

NO.	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1.	Startup Channel-Low Count Rate	$\geq$ 4.2 cps	$\geq$ 3.2 cps
2a.	Linear Channel-Low Power RWP (Channels 3, 4 and 5)	$\geq$ 5% Indicated Thermal Power (h)	$\geq$ 5% Indicated Thermal Power (h)
2b.	Linear Channel-Low Power RWP (Channels 6, 7 and 8)	$\geq$ 5% Indicated Thermal Power (h)	$\geq$ 5% Indicated Thermal Power (h)
3a.	Linear Channel-High Power RWP (Channels 3, 4 and 5)	$\leq$ 30% Indicated Thermal Power (i)	$\leq$ 30% Indicated Thermal Power (i)
3b.	Linear Channel-High Power RWP (Channels 6, 7 and 8)	$\leq$ 30% Indicated Thermal Power (i)	$\leq$ 30% Indicated Thermal Power (i)

Notes are on Pages 3/4 3-14 and 3/4 3-15

TABLE 3.3.1-4 (Part 2)

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

NO.	FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS
1.	Startup Channel - Low Count Rate	2	1	Above 1.0E-03% RATED THERMAL POWER
2a.	Linear Channel - Low Power RWP (Channels 3, 4, and 5)	2	1	(e)
2b.	Linear Channel - Low Power RWP (Channels 6, 7, and 8)	2	1	(e)
3a.	Linear Channel - High Power RWP (Channels 3, 4, and 5)	2 (d)	1	None
3b.	Linear Channel - High Power RWP (Channels 6, 7, and 8)	2 (d)	1	None

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Notes are on Pages 3/4 3-14 and 3/4 3-15

NOTES FOR TABLES 3.3.1-1 THROUGH 3.3.1-4

a) 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.

b) With one primary coolant high level moisture monitor tripped, TRIPS of either loop primary coolant moisture monitors will cause full scram. Hence, number of OPERABLE channels (1) minus minimum number required to cause scram (0) equals one, the minimum degree of redundancy.

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 inoperable channel must be in the tripped condition, unless the TRIP of the channel will cause the protective action to occur. Failure to TRIP the inoperable channel requires taking the appropriate ACTION as listed on Pages 3/4 3-1 and 3/4 3-2.

e) RWP bypass permitted if the bypass also causes associated single channel scram.

f) For loop monitors only, permissible bypass conditions include:
   I. Any circulator buffer seal malfunction.
   II. Loop hot reheat header high activity.

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

h) LOW POWER RWP bistable resets at 4% RATED THERMAL POWER after reactor initially exceeds 5% RATED THERMAL POWER.

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

j) Item 5a. must be accompanied by Item 5c. for Loop 1 shutdown. Item 5b. must be accompanied by Item 5c. for Loop 2 shutdown.

NOTES FOR TABLES 3.3.1-1 THROUGH 3.3.1-4

k) Separate instrumentation is provided on each circulator for this functional unit.

l) A primary coolant Dewpoint Moisture Monitor shall not be considered OPERABLE unless the following conditions are met:

1)

Reactor Power Range	Minimum Sample Flow
Startup to 2%	1 scc/sec.
Greater than 2% - 5%	5 scc/sec.
Greater than 5% - 20%	15 scc/sec.
Greater than 20% - 35%	30 scc/sec.
Greater than 35% - 100%	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 degrees F.

4) Fixed alarms of 1 scc/sec and 75 scc/sec are OPERABLE.

TABLE 4.3.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF SCRAM SYSTEM

Channel Description	Function*	Frequency	Method
1. Manual (Control Room)	a. TEST	R	a. Manually TRIP system.
2. Manual (I-49)	a. TEST	M	a. Manually TRIP each channel.
3. Startup Channel	a. CHECK	D	a. Comparison of two separate channel indicators.
	b. TEST	P	b. Apply test signal to verify TRIPS and alarms.
	c. CALIBRATE	R	c. 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.
4. Linear Power Channel	a. CHECK	D	a. Comparison of 6 separate channel indicators.
	b. TEST	M	b. Apply test signal to verify trips and alarms.
	c. CALIBRATE	D	c. Channel adjusted to agree with heat balance calculation.
	d. CALIBRATE	R	d. Apply test signals to adjust TRIPS and indications.
5. Wide Range Power Channel	a. CHECK	D	a. Comparison of three separate indicators.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF SCRAM SYSTEM (Cont'd)

Channel Description	Function*	Frequency	Method
5. (Cont'd)	b. TEST	P	b. Apply test signals to verify TRIPS and alarms.
	c. CALIBRATE	M	c. Channel adjusted to agree with heat balance calculation.
	d. CALIBRATE	R	d. Apply test signals to adjust TRIPS and indications.
6. Primary Coolant Moisture (All Channels)	a. CHECK	D	a. Comparison of two separate high level channel mirror temperature indications.
	b. CHECK	D	b. Comparison of six separate low level channel mirror temperature indications.
	c. CALIBRATE	R	c. Inject moisture laden gas into sample lines.
	d. CHECK	D	d. Verification of eight separate monitor's sample flow, per Item (1) of Notes for Tables 3.3.1-1 through 3.3.1-4.
	e. TEST	M	e. Verify that each of the eight monitors will alarm on low and high sample flow.
7. Primary Coolant Moisture (High Level Channels)	a. TEST	M	a. TRIP one high level, one low level channel, pulse another low level channel.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING
OF SCRAM SYSTEM (Cont'd)

Channel Description	Function*	Frequency	Method
8. Reheat Steam Temperature	a. CHECK	D	a. Comparison of the averaged thermocouple channel input indications.
	b. TEST	M	b. TRIP channel, verify alarms and indications. Apply test signal to verify TRIPS and alarms.
	c. CALIBRATE	R	c. Compare each thermocouple output to an NBS traceable standard. Apply test signal to adjust TRIPS and indicators.
9. Primary Coolant Pressure	a. CHECK	D	a. Comparison of six separate channel indicators.
	b. TEST	M	b. TRIP channel, apply test signal to verify TRIPS and alarms.
	c. CALIBRATE	R	c. Known pressure applied to sensor. Apply test signal to adjust TRIPS and indicators.
10. Circulator Inlet Temperature	a. CHECK	D	a. Comparison of eight separate indicators.
	b. TEST	M	b. TRIP channel, apply test signal to verify TRIPS and alarms.
	c. CALIBRATE	R	c. Compare each thermocouple output to an NBS traceable standard. Apply test signal to adjust TRIPS and indicators.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK,
CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF SCRAM SYSTEM (Cont'd)

Channel Description	Function*	Frequency	Method
11. Hot Reheat Header Pressure	a. TEST	M	a. Reduce pressure at sensor to TRIP channel, verify alarms and indications.
	b. CALIBRATE	R	b. Known pressure applied at sensor to adjust TRIPS.
12. Main Steam Pressure	a. TEST	M	a. Reduce pressure at sensor to TRIP channel, verify alarms and indications.
	b. CALIBRATE	R	b. Known pressure applied at sensor to adjust TRIPS.
13. Two Loop Trouble	a. TEST	M	a. Special test module used to TRIP channel by energizing each of four appropriate pairs of two-loop trouble relays.
	b. TEST	R	b. TRIP logic to cause two loop trouble scram.
14. Plant 480 V Power Loss	a. TEST	M	a. TRIP each channel by applying simulated loss of voltage signal; verify alarms and indications.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF SCRAM SYSTEM (Cont'd)

Channel Description	Function*	Frequency	Method
15. High Reactor Building Temperature (Pipe Cavity)	a. CHECK	D	a. Comparison of three separate channel indicators.
	b. TEST	M	b. TRIP channel, verify alarms and indications. Apply test signal to verify TRIPS and alarms.
	c. CALIBRATE	R	c. Compare each thermocouple output to an NBS traceable standard to adjust temperature TRIP point.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF LOOP SHUTDOWN SYSTEM

Channel Description	Function*	Frequency	Method
1. Circulator 1A and 1B Tripped	a. TEST	M	a. Pulse test and verify proper indications.
	b. TEST	R	b. TRIP both circulators to test loop shutdown.
2. Circulator 1C and 1D Tripped	a. TEST	M	a. Pulse test and verify proper indications.
	b. TEST	R	b. TRIP both circulators to test loop shutdown.
3. Steam Generator Penetration Pressure	a. TEST	M	a. Pressure switches actuated by pressure applied.
	b. TEST	M	b. Pulse test each channel with another channel tripped and verify proper indications.
	c. CALIBRATE	R	c. Known pressure applied at sensor to adjust TRIP.
4. Reheat Header Activity	a. CHECK	D	a. Comparison of three separate indicators in each loop.
	b. TEST	M	b. Pulse test each channel with another channel tripped and verify proper indications.
	c. CALIBRATE	R	c. Expose sensor to known radiation source and adjust TRIPS and indicators.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-2

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF LOOP SHUTDOWN SYSTEM (Cont'd)

Channel Description	Function*	Frequency	Method
5. Superheat Header Temperature	a. CHECK	D	a. Comparison of 3 separate temperature indicators per loop.
	b. CHECK	D	b. Comparison of 3 separate temperature differential indicators.
	c. TEST	M	c. Pulse test one channel with another channel tripped and verify proper indications.
	d. CALIBRATE	R	d. Compare each thermocouple output to an NBS traceable standard. Apply test signal to adjust TRIPS and indicators.
6. Primary Coolant Moisture (Low Level Channels)	a. TEST	M	a. TRIP each channel, verify proper indications.
	b. TEST	M	b. TRIP each channel, pulse test other loop to check loop identification.
7. Primary Coolant Pressure	a. TEST	M	a. Pulse test one channel with another channel tripped and verify proper indications, both channels.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF CIRCULATOR TRIP SYSTEM

Channel Description	Function*	Frequency	Method
1. Circulator Speed-Steam and Water	a. CHECK	D	a. Comparison of 6 separate speed indications per circulator.
	b. TEST	M	b. Apply test signal to verify TRIP setting and indicators.
	c. TEST	M	c. Pulse test one channel with another channel tripped, and verify proper indications.
	d. CALIBRATE	R	d. Known pulse frequency applied at sensor to adjust TRIPS and indicators.
2. Feedwater Flow	a. CHECK	D	a. Comparison of 6 separate indicators per loop.
	b. TEST	M	b. Apply test signal to verify TRIP setting and indications.
	c. TEST	M	c. Pulse test one channel with another channel tripped, and verify proper indications.
	d. CALIBRATE	R	d. Apply known delta P at flow transmitter. Apply test signal to adjust TRIPS and indicators.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

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TABLE 4.3.1-3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF CIRCULATOR TRIP SYSTEM (Cont'd)

Channel Description	Function*	Frequency	Method
3. Circulator Bearing Water Pressure	a. CHECK	D	a. Comparison of 3 separate indicators/ circulator.
	b. TEST	M	b. Pulse test one channel with another channel tripped and verify proper indications.
	c. CALIBRATE	R	c. Known pressure applied to adjust TRIP setting.
4. Circulator Penetration Pressure	a. TEST	M	a. Pressure switches actuated by pressure applied.
	b. TEST	M	b. Pulse test one channel with another channel tripped and verify proper indications.
	c. CALIBRATE	R	c. Known pressure applied at sensor to adjust TRIP setting.
5. Circulator Drain Pressure	a. CHECK	D	a. Comparison of 3 separate indicators/ circulator.
	b. TEST	M	b. Pulse test one channel with another channel tripped and verify proper indications.
	c. CALIBRATE	R	c. Known pressure applied at sensor to adjust TRIP setting.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING
OF CIRCULATOR TRIP SYSTEM (Cont'd)

Channel Description	Function*	Frequency	Method
6. Circulator Seal Malfunction	a. CHECK	D	a. Comparison of 3 separate indicators/ circulator.
	b. TEST	M	b. Pulse test one channel with another channel tripped and verify proper indications.
	c. CALIBRATE	R	c. Known pressure applied at sensor to adjust TRIP setting.
7. Circulator Trip (Manual)	a. TEST	R	a. TRIP steam turbine drives. Verify water turbine automatic start.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK,
CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF CIRCULATOR TRIP SYSTEM (Cont'd)

NO.	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	ACTUATION LOGIC TEST	APPLICABILITY
8a.	Steam Leak Detection Turbine Building Loop 1	D	R(a)	R(a)	R(b)	STARTUP*, LOW POWER, POWER
8b.	Steam Leak Detection Reactor Building Loop 1	D	R(a)	R(a)	R(b)	STARTUP*, LOW POWER, POWER
8c.	Steam Leak Detection Turbine Building Loop 2	D	R(a)	R(a)	R(b)	STARTUP*, LOW POWER, POWER
8d.	Steam Leak Detection Reactor Building Loop 2	D	R(a)	R(a)	R(b)	STARTUP*, LOW POWER, POWER

Notes to items 8a. through 8d. above only:

(a) The CHANNEL CALIBRATION/CHANNEL FUNCTIONAL TEST consists of verifying the rate of rise setpoint and checking for opens and shorts in the sensor cable.

(b) The SLRDIS Detection and Logic racks shall be verified to have a response time less than or equal to 7.1 seconds when a simulated rate-of-rise trip input signal is used to actuate the output relay logic.

* Applicable only above 2% RATED THERMAL POWER

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TABLE 4.3.1-4

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF ROD WITHDRAWAL PROHIBIT SYSTEM

Channel Description	Function*	Frequency	Method
1. Start-up Channel	a. CHECK	D	a. Comparison of two separate channel indicators.
	b. TEST	P	b. Apply test signal to verify all TRIPS and alarms.
	c. CALIBRATE	R	c. The 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.
2. Linear Channel	a. CHECK	D	a. Comparison of 6 separate level indicators.
	b. TEST	M	b. Apply test signal to verify TRIPS and alarms.
	c. CALIBRATE	D	c. Channel adjusted to agree with heat balance calculation.
	d. CALIBRATE	R	d. Apply test signals to adjust TRIPS and indications.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

TABLE 4.3.1-4

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF ROD WITHDRAWAL PROHIBIT SYSTEM (Cont'd)

Channel Description	Function*	Frequency	Method
3. Wide Range Power Channel	a. CHECK	D	a. Comparison of three separate indicators.
	b. TEST	P	b. Apply test signals to verify TRIPS and alarms.
	c. CALIBRATE	M	c. Channel adjusted to agree with heat balance calculation.
	d. CALIBRATE	R	d. Apply test signals to adjust TRIPS and indications.
4. Multiple Rod Pair Withdrawal	a. TEST	P	a. Attempt two rod pair withdrawal.
	b. CHECK	R	b. Simulate current through sensor to verify TRIPS and alarms.

* TEST = CHANNEL FUNCTIONAL TEST, CHECK = CHANNEL CHECK, CALIBRATE = CHANNEL CALIBRATION

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. Some protective actions are necessary only during STARTUP and require bypass at LOW POWER and POWER; others are required during POWER and are bypassed at STARTUP and LOW POWER. A simple method, based on a minimum of administrative control, has been devised to sequence and bypass protective actions. The equipment consists of two selector switches (Reactor Mode and Interlock Sequence) on the reactor control board. This specification provides the limiting conditions for operation necessary to preserve the effectiveness of these instrument systems.

For the limiting conditions for the PLANT PROTECTIVE SYSTEM instrumentation shown on Tables 3.3.1-1 through 3.3.1-4, the following definition applies:

> Degree of Redundancy - Difference between the number of OPERABLE channels and the minimum number of OPERABLE channels which when tripped will cause an automatic system TRIP.

If the minimum OPERABLE channels or the minimum degree of redundancy for each functional unit of a table cannot be met or cannot be bypassed under the stated permissible bypass conditions, the associated ACTION statements apply.

If, within the indicated time limit, the minimum number of OPERABLE channels and the minimum degree of redundancy can be re-established, the system is considered normal and no further ACTION needs to be taken.

The TRIP level settings are included in this section of the specification. The BASES for these settings are briefly discussed below. Additional discussions pertaining to the Rod Withdrawal Prohibit, Scram, Loop Shutdown and Circulator Trip inputs may be found in FSAR Sections 7.1.2.2, 7.1.2.3, 7.1.2.4 and 7.1.2.6, respectively. High moisture instrumentation is discussed in FSAR Section 7.3.2.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which TRIP SETPOINTS can be measured and calibrated, ALLOWABLE VALUES and TRIP SETPOINTS have been specified in Part 1 of Tables 3.3.1-1 through 3.3.1-4. The methodology used for calculating the ALLOWABLE VALUES and TRIP SETPOINTS is discussed in Technical Specification LSSS 2.2.1.

BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

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 PLANT 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, preoperational testing, or other low power operations.

Linear Channel - High (Neutron Flux)

See Technical Specification LSSS 2.2.1.

Primary Coolant Moisture - High

See Technical Specification LSSS 2.2.1.

Reheat Steam Temperature - High

See Technical Specification LSSS 2.2.1.

Primary Coolant Pressure - Programmed Low

See Technical Specification LSSS 2.2.1.

Primary Coolant Pressure - Programmed High

See Technical Specification LSSS 2.2.1.

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 reactor 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.

BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

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 this case. In addition, the superheater outlet stop check valves are automatically closed to reroute main steam to the flash tank (through the individual loop bypass valves and desuperheaters). This is required 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 - 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% RATED THERMAL POWER may continue following the shutdown of the other loop (unless preceded by a 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. However, actuation of both Steam Line Rupture Detection/Isolation System (SLRDIS) loops, effectively shuts down both loops because it sends an actuation logic signal to all four circulator TRIP logic channels. The consequences of a two-loop shutdown and subsequent loss of forced circulation have been analyzed and found to be acceptable. The consequences are bounded by an interruption of forced circulation cooling accident described in FSAR Section 14.4.2.2, SAFE SHUTDOWN COOLING.

High Reactor Building Temperature (Pipe Cavity)

High temperature in the pipe cavity would indicate the presence of a steam leak. A 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. The setpoint has been set above the SLRDIS pre-trip temperature alarm.

Loop Shutdown Inputs

The 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.

BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

Shutdown of Both Circulators (Loop Shutdown Logic)

Shutdown of both circulators in one loop is a loop shutdown input so that secondary coolant flow is automatically isolated to the affected loop's steam generator upon loss of PRIMARY COOLANT FLOW in that loop. This loop shutdown ensures proper PLANT PROTECTIVE SYSTEM action through the two-loop trouble scram in the event of the loss of all four circulators. Low feedwater flow to both loops can result in automatic TRIP of all four circulators, which would activate the two-loop trouble scram.

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 released, the steam generator contents are also dumped.

The steam generator interspace rupture discs are set at 825 psig (nominal). The burst pressure range (plus or minus 2%) is 808 psig to 842 psig (Technical Specification LSSS 2.2.1, Table 2.2.1-1). 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.

High Reheat Header Activity - (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 ensures detection of major reheat tube ruptures 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 High Differential Temperature Between Loop 1 and Loop 2

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

BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

Circulator Shutdown Inputs

All circulator shutdown inputs are equipment protection items. With the exception of Circulator Speed-High Water, all circulator shutdown inputs are connected to the two-loop trouble scram logic 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 Water is included to afford protection to the water turbine assembly against the effects of overspeed during continued core cooling upon loss of steam drive capability.

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 rate, 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.

Loss of Circulator Bearing Water

In order to prevent circulator damage upon loss of normal and backup bearing water supplies, a gas pressurized 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 (plus or minus 2%) is 808 psig to 842 psig (Technical Specification LSSS 2.2.1, 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).

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BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

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 greater than or equal to 8.5 psid, will initiate an automatic shutdown of the circulator.

Circulator Speed - High Steam

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

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 required.

Circulator Seal Malfunction (Low/High)

A high reverse differential of -6.1" w.g. would be reasonable evidence that bearing water is leaking into the primary coolant system. An increasing differential pressure of +76.1" w.g. 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.

BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

Steam Leak Detection in the Reactor Building

Steam Leak Detection in the Reactor Building is required for equipment qualification of SAFE SHUTDOWN COOLING Systems. The ALLOWABLE VALUE is set at ≤ 52.8 degrees F per minute rate of rise in order to prevent exceeding the harsh environment temperature profile to which the safe shutdown electrical equipment is qualified, per the requirements of 10CFR50.49. A setpoint calculation analysis performed per ISA Standard S67.04-1982 and RG 1.105 Rev. 2, results in the stated ALLOWABLE VALUE and TRIP SETPOINT as specified in the LCO and this basis. The TRIP SETPOINT has been established with sufficient margin between the technical specification limit for the process variable and the nominal TRIP SETPOINT to allow for 1) inaccuracy of the instruments; 2) uncertainties in the calibration; 3) instrument drift that could occur during the interval between calibrations; and 4) inaccuracies due to ambient temperature changes, vibration and other environmental conditions. The TRIP SETPOINT is set at ≤ 52.3 degrees F per minute rate of rise until such time as the drift characteristics of the detection system are better understood from actual plant operating experience and the assumptions used in the setpoint analysis are verified.

SLRDIS design incorporates two panels, each with its own set of sensors for the reactor and turbine buildings and dual logic trains in each panel. The SLRDIS design preserves the single failure concept. A single failure will neither cause nor prevent SLRDIS actuation in the event of a high energy line break. The probability of an inadvertent actuation is extremely small due to the matrix logic employed for circulator TRIP and valve actuation. The SLRDIS panels are referred to as "loops"; however, due to the way the outputs of the panels are combined to provide protective action and satisfy the single failure concept, the SLRDIS loops do not correspond to primary or secondary loops.

For each SLRDIS loop, the OPERABILITY requirements and their respective ACTIONs represent good operating practices and judgment for a four channel detection system with a 2 of 4 coincidence TRIP logic. The fourth channel may be placed in bypass for test and/or maintenance purposes, subject to the ACTION statement restrictions, while preserving a 2 of 3 coincidence logic OPERABLE. The Steam Line Rupture Detection/Isolation System as designed and installed has spare channels available for input. Any of the available channels may be selected for input signal processing provided the surveillances are current on the channels used. The SLRDIS is required to be OPERABLE only above 2% RATED THERMAL POWER. Analyses at 2% RATED THERMAL POWER demonstrate that automatic actuation of SLRDIS is not likely to occur during a high energy line break lasting until it is manually terminated at one hour following initiation. The temperatures as analyzed in both the reactor and turbine buildings stay well below the temperature for which the equipment is qualified.

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BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

The ACTION statements for inoperable SLRDIS detection and information processing equipment allow one channel in each building to be inoperable for up to 7 days; a second inoperable channel in either building requires that power be reduced to below 2% within 12 hours. The 7 day ACTION time for a single detector channel is acceptable based on preservation of a 2 out of 3 coincidence detection system still in operation. ACTION 3 is applicable to other functions within the SLRDIS instrumentation panel such as loss of power from instrument buses, or other failures in the logic trains and associated electronics. A 12-hour time period in ACTION 3 for inoperability of those associated SLRDIS functions minimized the time that SLRDIS may operate with limited functional capability. An inoperable valve or associated equipment is allowed for 72 hours. High energy line break analysis for environmental qualification assumes the worst-case single active failure. Thus, a single valve inoperable for up to 72 hours is within the bounds of analysis. When two or more valves and/or associated equipment are inoperable, 24 hours is allowed to restore the inoperable equipment. Repairs may be performed while the plant is at greater than 2% RATED THERMAL POWER, thus minimizing thermal cycling of plant and installed equipment.

Steam Leak Detection in the Turbine Building is required for equipment qualification of SAFE SHUTDOWN COOLING Systems. Thus, the limits and BASIS are the same as discussed in the BASIS for steam leak detection in the reactor building.

Rod Withdrawal Prohibit Inputs

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

Startup Channel - Low Count Rate

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

Linear Channel - Low Power RWP

Linear Channel (5% RATED THERMAL POWER) directs the reactor operator's attention to either a downscale failure of a power range channel or improper positioning of the Interlock Sequence Switch. (FSAR Sections 7.1.2.2 and 7.1.2.8)

Linear Channel - High Power RWP

Linear Channel (30% RATED THERMAL POWER) is provided to prevent control rod pair withdrawal if reactor power exceeds the Interlock Sequence Switch limit for LOW POWER. (FSAR Sections 7.1.2.2 and 7.1.2.8)

BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

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 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 through 4.3.1-4 are divided into three categories: 1) passive type indicating devices that can be compared with like units on a continuous basis; 2) semiconductor devices and detectors that may drift or lose sensitivity; and 3) on-off sensors which must be tripped by an external source to determine their setpoint. Drift tests by GA 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.

The surveillance requirements for the Steam Line Rupture Detection/Isolation System instrumentation in Table 4.3.1-3 include provisions for CHANNEL CHECK, CHANNEL CALIBRATION, CHANNEL FUNCTIONAL TEST and an ACTUATION LOGIC TEST. The frequency of CHANNEL CALIBRATION, at least once per REFUELING CYCLE, not to exceed 18 months, is consistent with the interval for testing and calibrating similar detectors (heat sensitive cabling used for fire detection). The manufacturer of the instrumentation recommends an 18 month interval for test/calibration of the electronics portion of the Steam Line Rupture Detection/Isolation System, thus, the CHANNEL FUNCTIONAL TEST is specified for that interval. The ACTUATION LOGIC TEST verifies proper operation of the SLRDIS Detection and Logic Racks from a simulated rate-of-rise input signal through and including actuation of the output logic relays. Time response of the SLRDIS Detection and Logic Racks is verified to be equal to or less than 7.1 seconds as assumed in the high energy line break analysis. The potential for an inadvertent actuation during testing suggests that logic testing be performed only when the plant is in SHUTDOWN. Thus, the surveillance requirements are specified for REFUELING but not to exceed 18 months. The SLRDIS control unit includes a supervision system that continuously and automatically monitors critical circuitry and internal components, and alarms SLRDIS trouble conditions to the operators.

Tests and calibrations of instrument channels in Tables 4.3.1-1 through 4.3.1-4 may be performed with either internal or external test signals. Use of the internal test signal is preferred, while equivalent external test signals are equally acceptable.

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INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

ANALYTICAL MOISTURE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.2.1 The following analytical moisture monitors shall be OPERABLE:

a. Upon entry into and operation in STARTUP from SHUTDOWN, two analytical moisture monitors (or alternate PPS dewpoint moisture monitor(s) placed in the "Indicate" mode), and

b. Upon entry into and operation in STARTUP from LOW POWER, one analytical moisture monitor (or alternate a PPS dewpoint moisture monitor placed in the "Indicate" mode).

APPLICABLILITY: STARTUP

ACTION:

a. Upon entry into and operation in STARTUP from SHUTDOWN:

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

2. With no moisture monitors* OPERABLE:

a) Restore one monitor to OPERABLE status or be in SHUTDOWN or LOW POWER within the next 90 minutes, and

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

b. Upon entry into 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 90 minutes.

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* A PPS dewpoint moisture monitor placed in the "Indicate" mode can be utilized to meet the intent of Specification 3.3.2.1 for an OPERABLE analytical moisture monitor.

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SURVEILLANCE REQUIREMENTS

4.3.2.1 The analytical moisture monitors shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL CHECK within 12 hours of entering STARTUP from LOW POWER,

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

c. Performance of a CHANNEL CHECK at least once per 48 hours while operating in STARTUP, (at least once per 4 hours while relying on the PPS dewpoint monitor(s)),

d. Verifying the sampling line flow rate is between 250 and 2500 scc/minute, at least once per 7 days, when the analytical moisture monitors are in use,

e. Verifying that the sampling line heat tracing is energized, at least once per 7 days when the analytical moisture monitors are in use, and

f. Performance of a CHANNEL CALIBRATION at least once per REFUELING CYCLE.

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BASIS FOR SPECIFICATION LCO 3.3.2.1/SR 4.3.2.1

While the plant is in STARTUP, analytical moisture monitors are required as backup to the PPS moisture monitors for administration of Specification 3.4.3. One moisture monitor is sufficient to detect primary coolant moisture content on a continual basis. The analytical moisture monitor sample line outside of the instrument penetration is heat traced to ensure that the limiting temperature for the sample line will occur in the PCRV wall. This ensures that a coolant sample will be delivered to the instruments without loss of moisture due to condensation in the sample line. A description of the moisture monitors can be found in Section 7.3.2 of the FSAR.

Two analytical 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 (other than an alarm function). Alternative moisture monitors can also be placed in service sampling primary coolant, such as through realignment of a moisture monitor in the analytical system or utilization of OPERABLE (as defined in Specification 3.3.1) PPS dewpoint moisture monitors in the "Indicate" mode. The utilization of dewpoint moisture monitors is acceptable to meet the intent of Specification 3.3.2.1. (Note that in the "Indicate" mode, a trip is input to the PPS channel). Operator action may be required 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 because the fuel graphite oxidation rate is low. As indicated by Figure 4.2 in Document GA-A13677, "Test and Evaluation of the FSV Dew Point Moisture Monitors System", one of the limiting parameters for determining required response times to shut down the reactor 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 5% RATED THERMAL POWER, response time to scram the reactor to limit oxidation to 1% by weight is approximately 6700 seconds.

The 90 minute ACTION times are based on the time required to limit graphite oxidation to 1%. This also minimizes the amount of metal corrosion and oxidation of burnable poison materials.

The SURVEILLANCE INTERVAL specified for demonstrating OPERABILITY and for calibration of this instrumentation will ensure the proper operation of these detectors.

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 TRIP 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 3.0.3 and 3.0.4 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 for the OPERATIONAL MODES and at the frequencies shown in Table 4.3.2-1.

TABLE 3.3.2-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Process Monitors				
a. Steam/Water Dump Tank Monitors				
Radiation Shine (RT-93250-12, RT-93251-12)	1	P,L	≤ 3.0 mR/hr	2
b. PCRV Relief Valve Piping Monitor				
Radiation Shine (RT-93252-12)	1	P,L,S/U	≤ 3.0 mR/hr	2
2. Accident Monitors				
a. Reactor Building Accident Monitor				
Refueling Floor-East Wall (RT-93250-14)	1	All	≤ 3.0 R/hr	3
b. Reactor Plant Exhaust Filter Monitor (RT-93251-1)	1	All	≤ 2.0 R/hr	3
c. Criticality Alarm for the New Fuel Storage Building	1	When new fuel is being stored in the New Fuel Storage Building	< 20 mR/hr	1

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 next 36 hours.

ACTION 3 - With the number of OPERABLE channels less than the minimum channels OPERABLE requirement, initiate an alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

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TABLE 4.3.2-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	APPLICABLE MODES
1. 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
2. Accident Monitors				
a. Reactor Building Accident Monitor (RT-93250-14)	D	R	M	All
b. Reactor Plant Exhaust Filter Monitor (RT-93251-1)	D	R	M	All
c. Criticality Alarm for the New Fuel Storage Building	W	A	M	When new fuel is being stored in the New Fuel Storage Building

BASIS FOR SPECIFICATION LCO 3.3.2.2/SR 4.3.2.2

The OPERABILITY of the radiation monitoring channels ensures that: a) the radiation levels are measured in the areas served by the individual channels, and b) an alarm is initiated when the radiation level TRIP SETPOINT is exceeded.

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*. Additionally, accident monitors to meet the intent of NUREG 0737, Item II.F.1.3, are included in Table 3.3.2-1. The refueling floor-east wall area radiation monitor also serves as an accidental criticality monitor (RT-93250-14). The refueling floor-east wall monitor and the reactor plant exhaust filter monitor (RT-93251-1) may be used for Design Basis Accident Number 2, Rapid Depressurization of the Primary Coolant System. When new fuel is stored in the new fuel storage building, a criticality alarm is located in the building.

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

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

    1. 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.

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* While FSAR Table 7.3-2 is titled Process and Area Radiation Monitoring Systems, the monitors are herein functionally considered to all be process monitors.

BASIS FOR SPECIFICATION LCO 3.3.2/SR 4.3.2 (Continued)

2. 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).

    a) Ventilation exhaust monitors - RT-7324-1,-2, RT-7325-1,-2, RT-73437-1,-2, RT-4801, RT-4802, RT-4803

    b) Gas waste header exhaust - RT-6314-1,-2

    c) Secondary coolant air ejector- RT-31193

3. Radioactive liquid effluent monitoring includes the following, for which the requirements are given in Specific tion 8.1.2 and Specification 8.1.3.

    a) Radioactive liquid waste discharge - RT-6212, RT-6213

    b) Gas waste compressor cooling activity - RT-46211, RT-46212

4. The secondary coolant reheat steam piping monitors (RT-93250-10,-11; RT-93251-10,-11; and RT-93252-10,-11) are included as part of the PPS loop shutdown (Specification 3.3.1).

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

b. The accident monitoring instruments included in Table 3.3.2-1 involve the high range reactor building radiation monitor (RT-93250-14), the reactor plant exhaust filter monitor (RT-93251-1), and the criticality alarm for the new fuel storage building.

The ACTION statements are consistent for comparable instrumentation in the LWR Standard Technical Specifications.

The SURVEILLANCE INTERVAL specified for CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION conform to industry practice and the SURVEILLANCE INTERVALS given in the Standard Technical Specifications for LWRs and are therefore considered adequate to ensure the proper operation of these detectors.

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 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 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 Following a seismic event, all of the actuated seismic monitoring instruments shall be restored to OPERABLE status within 24 hours and the calibration of the vertical seismic triggers shall be checked via a CHANNEL FUNCTIONAL TEST within 5 days following the seismic event. For all seismic instruments found out of calibration, a CHANNEL CALIBRATION shall be performed within 30 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. Upon the actuation of a seismic monitoring instrument due to a seismic event, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon unit features important to safety.

TABLE 3.3.2-2

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	SETPOINT	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs and Vertical Seismic Triggers			2
a. PCRV Support Ring*	$\leq$ 0.015g	0 to 1g	
b. Top of PCRV*	$\leq$ 0.015g	0 to 1g	
c. Visitors Center	$\leq$ 0.015g	0 to 1g	
2. Seismoscopes			2
a. PCRV Support Ring	N/A		
b. Top of PCRV	N/A		
c. Visitors Center	N/A		

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* With control room alarm

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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 and Vertical Seismic Triggers			
a. PCRV Support Ring**	Q*	R	SA
b. Top of PCRV**	Q*	R	SA
c. Visitors Center	Q*	R	SA
2. Seismoscopes			
a. PCRV Support Ring	M	R	N/A
b. Top of PCRV	Q	R	N/A
c. Visitors Center	M	R	N/A

---

* Except seismic trigger

** With control room alarms

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, the response of the facility, and to sound an alarm in the event that a disturbance greater than the setpoint is experienced. This capability permits comparison of the measured response to that used in the design basis for the facility to determine if reactor shutdown or inspection is necessary pursuant to FSAR Section 7.3.9, and the plant emergency procedures.

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

The ACTION statements and SURVEILLANCE INTERVALS specified for testing and calibration of the seismic instrumentation are consistent with the Standard Technical Specifications for LWR's and industry practice and are, therefore, considered adequate to ensure the instruments operate as intended.

A CHANNEL CHECK will be performed once per 92 days to verify the OPERABILITY of the battery pack and charger. 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 verified on-site. This on-site calibration verification will also be performed following seismic events which cause the seismic triggers to be actuated. Seismic triggers so determined to be out of calibration will be sent to the manufacturer for CHANNEL CALIBRATION. All other seismic monitoring instrumentation found out of calibration can be calibrated on-site.

The seismoscopes are smoked glass devices wherein vibration causes a needle to etch a trace in the smoked glass. A CHANNEL CHECK consists of shining a flashlight through a viewing port to see if the device has been actuated. For readily accessible seismoscopes, this is performed monthly. The less accessible seismoscope on top of the PCRV is checked quarterly.

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 spectra data. Response spectrum data, when deemed necessary, will be obtained by off-site digitization of the film data from the accelerographs, and subsequent data reduction which requires several weeks. Included in this report will be the resultant effect of the seismic event on the Class I structures, systems, and components as listed in FSAR Table 1.4-1.

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 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.4 The meteorological monitoring instrumentation channels required in Table 3.3.2-3, shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3.2-3.

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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 NOAA 10 meter tower may be used as an acceptable alternate channel for these parameters.

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

* The daily CHANNEL CHECK only has to be performed when the NOAA 10 meter tower is used as an alternate channel.

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 reflects the guidance given in Proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants", September 1980.

Two meterological towers are installed approximately one-half mile north of the plant. The primary tower is 60 meters high and an alternate tower that provides data for the National Oceanic and Atmospheric Administration (NOAA 10 meter tower) is 10 meters high. The primary data for accident release dose assessment calculations utilizes wind speed at 10 meters, wind direction at 10 meters and delta temperature between 10 and 58 meters (for atmospheric stability). The remaining measurements serve as backups. The NOAA 10 meter tower may be used to meet the minimum OPERABLE channel requirement as shown on Table 3.3.2-3. CHANNEL CHECK and CHANNEL CALIBRATION frequencies applicable to the NOAA 10 meter tower are listed on Table 4.3.2-3.

The ACTION statement is consistent for comparable instrumentation in the LWR standard Technical Specifications.

The SURVEILLANCE INTERVALS specified for testing and calibration of the meteorological instrumentation ensure instrumentation OPERABILITY and are consistent with nuclear plant practice and recommendations given in Proposed Revision 1 to Regulatory Guide 1.23.

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

FIRE DETECTION AND ALARM SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.2.5 The fire detection instrumentation for each fire detection area shown in Table 3.3.2-4 shall be OPERABLE.

APPLICABILITY: At all times

ACTION: With the number of OPERABLE fire detection instrument(s) for a fire detection area 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 area(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 of 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, and

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.5.1 Each of the required fire detection instruments listed in Table 3.3.2.4 which are accessible during plant operation 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.

PECIFICATION SR 4.3.2.5 (Continued)

4.3.2.5.2 The fire detection supervised circuits associated with the detector alarms of each of the required fire detection instruments listed in Table 3.3.2-4 shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST 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 by performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days.

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TABLE 3.3.2-4

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS	MINIMUM INSTRUMENTS OPERABLE (Note 3)
1. Control Room	6	4
2. Auxiliary Electric Room	6	4 (Note 1)
Return Air Duct	1	1
3. 480 V Switchgear Room	6	4 (Note 1)
4. Reactor Bldg "J" Wall:		
Elevation 4756 to 4791	4	2
Elevation 4791 to 4829	4	2
Elevation 4829 to 4849	2	1
Elevation 4849 to 4881	2	1
5. Turbine Bldg "G" Wall:		
Elevation 4791 to 4811	2	1
Elevation 4811 to 4829	2	1
6. Refueling Floor HVAC Intake	1	1
7. Reactor Building HVAC Return Air Ducts	10	6
8. Building 10		
Switchgear Room & Ground Level	6	4 (Note 1)
Ground Level Under Mezzanine	5	4 (Note 1)
Battery Room	2	1 (Note 1)
9. Battery Room Exhaust Ducts	2	1
10. Hydraulic Valve Area (Level 6)	6	4
11. Hydraulic Power Unit (Level 1)	3 (Note 2)	1 Smoke or 1 Oil Mist Detector per Unit

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See footnote explanation on following page.

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TABLE 3.3.2-4 (Continued)

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS	MINIMUM INSTRUMENTS OPERABLE (Note 3)
12. Turbine Plant MCC2 & MCC3	2	1
13. Service Water Pump Building	2	1
14. Circulating Water Makeup Pump Building	2	1
15. Reactor Plant Exhaust Filter:		
Filter 1A	3	2 (Notes 1,4)
Filter 1B	3	2 (Notes 1,4)
Filter 1C	3	2 (Notes 1,4)
16. Diesel Generator Rooms:		
Room 1A	2	1 (Notes 1,4)
Room 1B	2	1 (Notes 1,4)

Notes:

1. Detector(s) automatically actuate fire suppression system(s).
2. Includes one oil mist detector which samples from both hydraulic power units.
3. Smoke detectors, unless otherwise indicated.
4. Heat detectors.

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. 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 watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABLE.

The smoke detection and alarm systems provide detection and alarm capability for the control room, auxiliary electric equipment 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.

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's 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 Fire Hazards Analysis dated November 13, 1978 (P-78182) forms the basis for installation of fire detectors.

The SURVEILLANCE INTERVAL specified for this instrumentation is consistent with industry practice for this type of equipment and ensures proper operation in the event of a fire.

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, with a setpoint adjusted to alarm at a chlorine concentration of 5 ppm or less.

APPLICABILITY: At all times, when chlorine gas is onsite for the purpose of chlorination.

ACTION:

a. With either the detection or alarm system inoperable:

1. Restore the system to OPERABLE status within 24 hours, or

2. Close the chlorine bottle discharge valves, except during chlorination. During chlorination, patrol the area once every 2 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.2.6 The chlorine detection and alarm system shall be demonstrated OPERABLE by:

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

b. At least once per 31 days, by performing a CHANNEL FUNTIONAL TEST, and

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

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.

Isolating the chlorine bottle discharge by closing the discharge valves, and patroling the area once every 2 hours during chlorination, will prevent an inadvertant release of chlorine if the chlorine detection and alarm system is inoperable.

The SURVEILLANCE INTERVAL specified for this instrumentation is consistent with LWR Standard Technical Specifications and ensures proper operation in the event of an accidental chlorine release.

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

POWER-TO-FLOW RATIO MONITORING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: POWER and LOW POWER*

ACTION: With the above required instrumentation inoperable:

a. Within 12 hours provide a backup means of recording data for computing the POWER-TO-FLOW RATIO, and restore the normal means of recording to OPERABLE status within 7 days, or

b. Be in a condition at which the specification does not apply within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable provided a backup means of recording data is in service for computing the POWER-TO-FLOW RATIO.

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per REFUELING CYCLE by performance of a CHANNEL CALIBRATION.

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* Applicable only above 15% RATED THERMAL POWER.

BASIS FOR SPECIFICATION LCO 3.3.2.7/SR 4.3.2.7

The POWER-TO-FLOW RATIO measurement system and recording instrumentation (XMS-11262-1 through XMS-11262-15, and XR-11262) provides an indication of the balance between the heat generation and removal within the primary coolant system. The POWER-TO-FLOW RATIO measurement system includes unique components, is not safety-related (not PPS, Class I, nor safe shutdown), and does not initiate any automatic protective system or control system actions.

The OPERABILITY of the 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 post-transient computer calculation or manual computation of the integrated time of the measured POWER-TO-FLOW RATIO transient and comparison with the allowable integrated time limits in Specification 2.1.1.

The Data Logger can provide backup for the strip-chart recorder when the strip-chart recorder is out of service. However, long-term operation with the Data Logger as the only POWER-TO-FLOW RATIO recorder is not desirable because the Data Logger is not controlled under the Technical Specifications. A 7 day ACTION time is acceptable for an inoperable POWER-TO-FLOW RATIO measurement system due to the availability of a backup means of monitoring the POWER-TO-FLOW RATIO and the fact that no automatic actions are initiated by the measurement system. Other parameters such as power, flow, temperatures, etc. are continuously monitored and will provide early indications of potential problems.

Operation for 12 hours without a means of recording data for computing the POWER-TO-FLOW RATIO is acceptable, since 12 hours of of operation at a POWER-TO-FLOW RATIO as abnormally high as 1.4 is acceptable, per Figure 2.1.1-1.

Verifying that the POWER-TO-FLOW RATIO is recorded on a daily basis during operation above 15% RATED THERMAL POWER is adequate to ensure the instrument is recording properly. In addition, any change in reactor power level should produce a change in the POWER-TO-FLOW RATIO indication. A CHANNEL CHECK daily, and CHANNEL CALIBRATION of the instrumentation on a once-per-REFUELING CYCLE basis, is acceptable by industry standards for this type of equipment.

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BASIS FOR SPECIFICATION LCO 3.3.2/SR 4.3.2 (Continued)

Applicability is limited to power levels above 15% RATED THERMAL POWER, in that the POWER-TO-FLOW RATIO recorder is only required in the range of 15% to 100% power to ensure 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 POWER-TO-FLOW-RATIO recorder is not required.

INSTRUMENTATION

3/4.3.2 MONITORING INSTRUMENTATION

CORE REGION OUTLET THERMOCOUPLES

LIMITING CONDITION FOR OPERATION

3.3.2.8 At least two thermocouples for each INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE indication shall be OPERABLE.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION:

a. With only one thermocouple OPERABLE in any region, operation may continue provided that within 24 hours a spare thermocouple is made OPERABLE within the refueling region, and can be monitored locally or in the control room.

b. With both of the thermocouples inoperable in any region:

   1. Restore at least one thermocouple to OPERABLE status within one hour, or

   2. Maintain the power and coolant flow in that region constant, and restore at least one thermocouple to OPERABLE status within 24 hours, or

   3. Be in at least SHUTDOWN within the next 24 hours.

c. The provisions of Specification 3.0.4 are not applicable for 7 days after reaching 2% RATED THERMAL POWER, to allow CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS

4.3.2.8 Each core region outlet thermocouple shall be demonstrated OPERABLE:

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

b. At least once per 92 days by performing a CHANNEL FUNCTIONAL TEST on the high temperature alarm, and

c. At least once per 18 months with the reactor power greater than 2% RATED THERMAL POWER and stable core temperatures, by performing a CHANNEL CALIBRATION. The provisions of Specification 4.0.4 are not applicable for 7 days after reaching 2% RATED THERMAL POWER.

BASIS FOR SPECIFICATION LCO 3.3.2.8/SR 4.3.2.8

The region outlet temperatures are determined by the region outlet thermocouples. These thermocouple measurements form the basis for adjusting the region flow orifices, and thus, the coolant flow through each refueling region. Together with the overall PRIMARY COOLANT FLOW, overall core power, orifice position, and control rod indicators they are used to ensure acceptable fuel temperatures during power operations. A complete description of the temperature measuring system as well as a discussion of the impact that failures within that system have on fuel temperatures is given in FSAR Section 3.6.7.

There are four independent thermocouples for each refueling region, each of which is capable of being used for temperature measurements. Two thermocouples provide simultaneous readouts in the control room and two are available as spares that can be made readable either locally or in the control room.

Any failure of the temperature measuring system is of little consequence, provided the status of the core power distibution or the core flow distribution is not changed. Short-term changes in the core power distribution can only be obtained by power level changes and/or anticipated or accidental control rod movement. Flow distribution changes can only be caused by changes in the orifice position. With the loss of all thermocouple readings from single regions or the total core, analytical data, as well as operating experience gained during power operation, are available to evaluate the power distribution in the reactor for a period of days, if the power level is not changed and control rod movements are kept to a minimum.

The complete loss of all temperature indication from a single region is highly improbable. Failure of one of these thermocouples leaves the operator the same flexibility for reactor operation as he has with both thermocouples operating and only temporarily reduces the reliability of the measurement. Therefore the specified ACTIONS and ACTION times within this specification are acceptable.

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BASIS FOR SPECIFICATION LCO 3.3.2/SR 4.3.2 (Continued)

Long term thermocouple drift is estimated to be less than or equal to 15 degrees F per year. 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 measurement. Daily CHANNEL CHECKS and an 18 month CHANNEL CALIBRATION are considered adequate since the expected drift in calibration is small and has been included in establishing Specification 3.2.2 (FSAR Section 7.3.3). The CHANNEL CALIBRATION is performed by locating a calibrated thermocouple within a calibration well of an outlet coolant thermometer assembly adjacent to the permanently installed thermocouple of any region covered by that assembly. Consequently, it can require up to 7 days for completion. In addition, the reactor power shall be greater than 2% with the core temperature stable (thermal equilibrium), so that the calibration is performed at near normal operating temperatures.

Each INDIVIDUAL REFUELING REGION OUTLET TEMPERATURE is equipped with an audible and visual high temperature alarm (FSAR Sections 7.3.3 and 3.6.7). A quarterly CHANNEL FUNCTIONAL TEST of this abnormal alarm is considered adequate since actual conformance with the temperature limits of Specification 3.2.2 is accomplished utilizing temperature indications on both the control panel and Data Logger. This test will be performed by manually adjusting the setpoint to actuate the alarm.

INSTRUMENTATION

3/4.3.3 THREE ROOM CONTROL COMPLEX TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3 The temperature of the control room, auxiliary electric equipment room, and the 480 volt switchgear room (the THREE ROOM CONTROL COMPLEX) shall not exceed 115 degrees F.

APPLICABILITY: At all times

ACTION: With one or more of the above areas in the THREE ROOM CONTROL COMPLEX exceeding 115 degrees F:

a. For more than 8 hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30 degrees F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the area inoperable.

SURVEILLANCE REQUIREMENTS

4.3.3 At least once per 24 hours, the temperature in each of the above areas shall be determined not to exceed 115 degrees F.

BASIS FOR SPECIFICATION LCO 3.3.3/SR 4.3.3

The limiting temperatures in the THREE ROOM CONTROL COMPLEX are established to ensure no overtemperature condition exists which could 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 ACTION statements are consistent for comparable instrumentation in the LWR standard Technical Specifications.

The SURVEILLANCE INTERVAL specified ensures adequate monitoring of temperatures in the designated plant areas.

PRIMARY COOLANT SYSTEM

3/4.4.1 PRIMARY COOLANT LOOPS AND COOLANT CIRCULATION

ABOVE 5% POWER

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two primary coolant loops shall be in operation and circulating primary coolant, each with:

a. Both the steam generator economizer-evaporator-superheater (EES) section and the reheater section operating (each section includes six modules), and

b. At least the minimum number of helium circulators operating to meet power level requirements as follows:

PERCENT RATED THERMAL POWER	MINIMUM NUMBER OF HELIUM CIRCULATORS
Greater than 50%	2 in each loop
Greater than 5% but less than or equal to 50%	1 in each loop

APPLICABILITY: POWER and LOW POWER

ACTION:

a. With only one primary coolant loop in operation, restore both loops to operating status within 12 hours or be in at least STARTUP within the next 24 hours.

b. With only one operating helium circulator in either loop, reduce reactor power to at least 50% RATED THERMAL POWER within 30 minutes.

c. With one steam generator section not operating in either primary coolant loop, restore all steam generator sections to operating status within 12 hours or be in at least STARTUP within the next 24 hours.

SPECIFICATION LCO 3.4.1.1 (Continued)

d. With no primary coolant loops in operation, be in at least SHUTDOWN within 10 minutes and restore forced circulation with at least one circulator in at least one loop within 90 minutes, or depressurize the PCRV in accordance with the applicable requirement below. If forced circulation is restored within 5 hours of initial loss, depressurization may be discontinued.

1. For reactor THERMAL POWER equal to or greater than 25% prior to SHUTDOWN, depressurize per Figure 3.4.1-1.

2. For reactor THERMAL POWER less than 25% prior to SHUTDOWN, depressurize per Figure 3.4.1-2.

3. With the reactor already SHUTDOWN, depressurize per Figure 3.4.1-3.

SURVEILLANCE REQUIREMENTS

4.4.1.1 At least once per 12 hours, the above required primary coolant equipment shall be verified to be in operation and circulating primary coolant.

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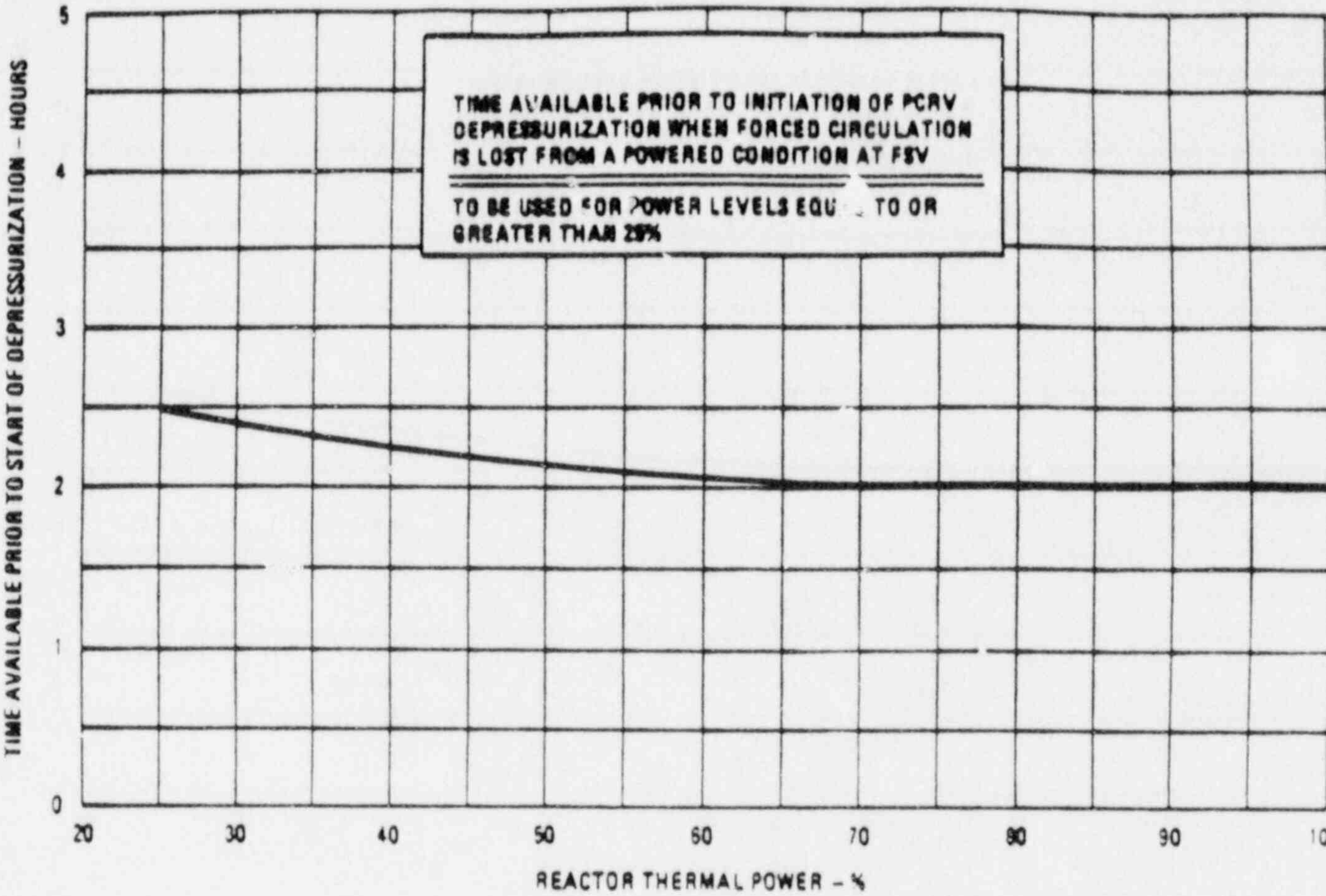

Time Available Prior to Initiation of PCRV Depressurization When Forced Circulation is Lost from a Powered Condition at FSV

Figure 3.4.1-1

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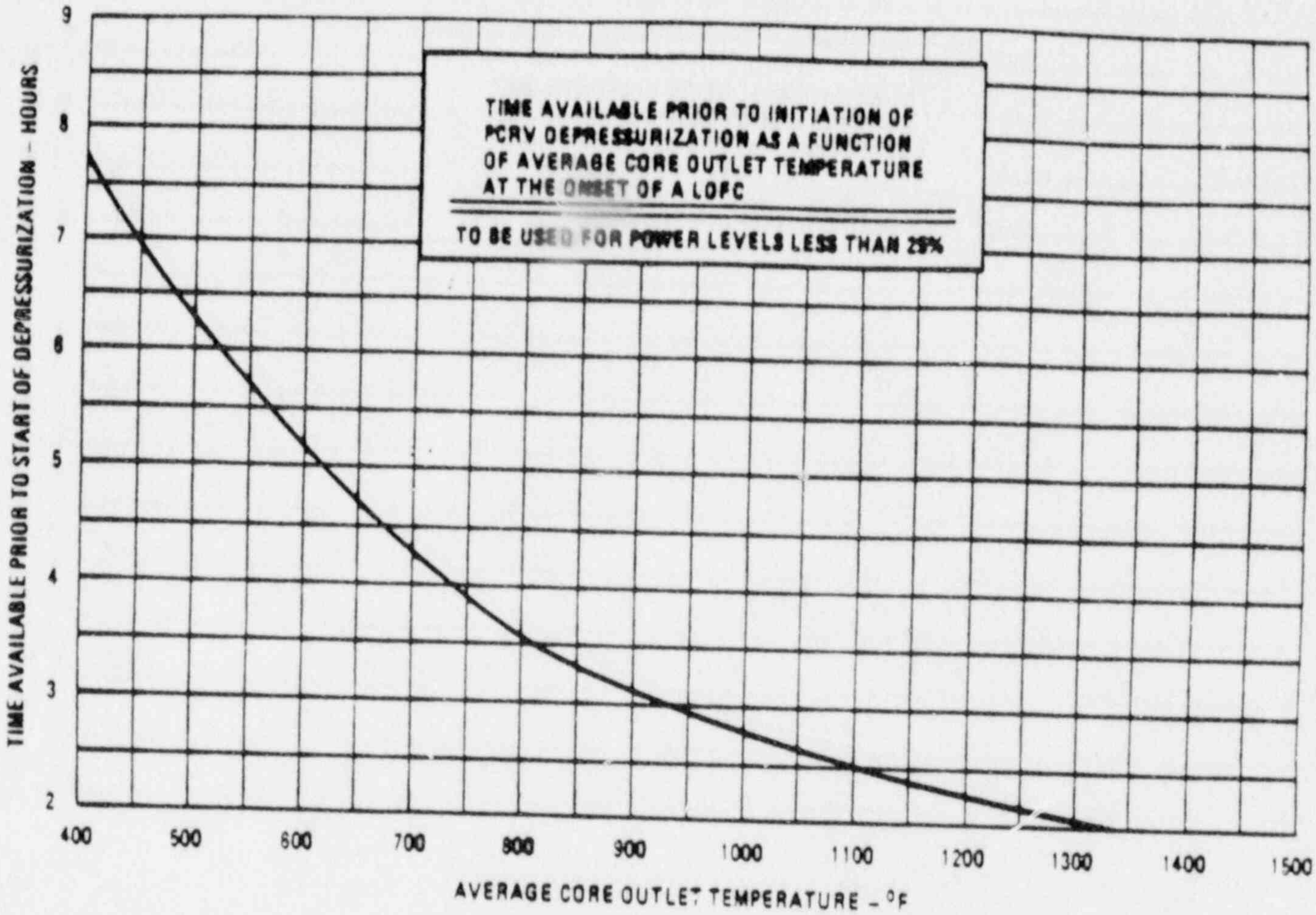

Time Available Prior to Initiation of PRCV Depressurization as a Function of Average Core Outlet Temperature at the Onset of a LOFC

Figure 3.4.1-2

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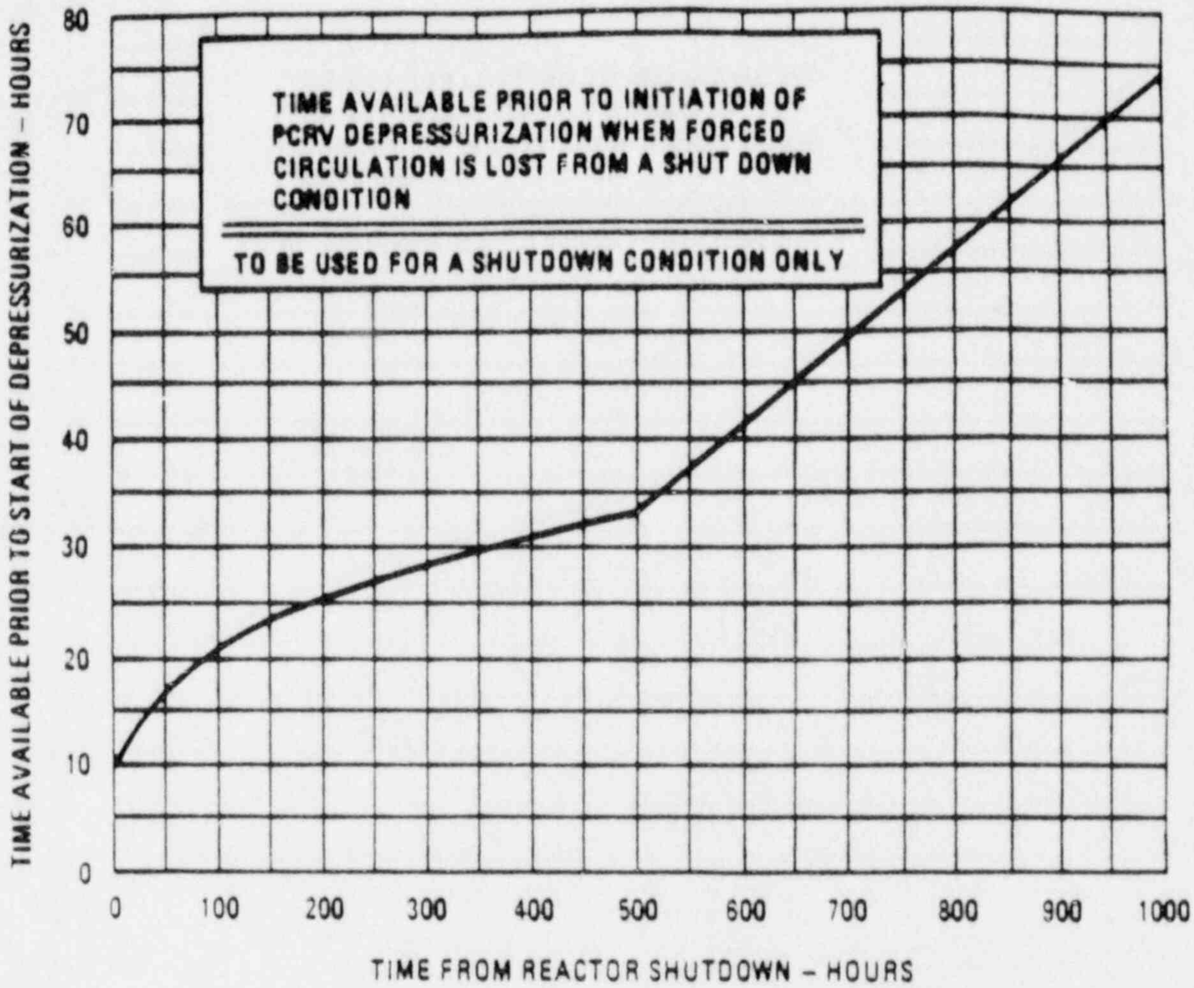

Time Available Prior to Initiation of PCRV Depressurization
When Forced Circulation is Lost from a Shut Down Condition

Figure 3.4.1-3

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PRIMARY COOLANT SYSTEM

3/4.4.1 PRIMARY COOLANT LOOPS AND COOLANT CIRCULATION

5% POWER AND BELOW

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least one primary coolant loop shall be in operation and circulating primary coolant, with at least:

a. One helium circulator operating, and

b. One steam generator section operating, either the economizer-evaporator-superheater (EES) or the reheater.

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING*

ACTION: a. With no operating primary coolant loops, be in at least SHUTDOWN within 10 minutes and suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes, and

b. Initiate PCRV depressurization in accordance with the time specified in Figures 3.4.1-2 or 3.4.1-3, as applicable.

SURVEILLANCE REQUIREMENTS

4.4.1.2 At least once per 12 hours, the above required primary coolant equipment shall be verified to be in operation and circulating primary coolant.

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* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

BASIS FOR SPECIFICATIONS LCO 3.4.1.1/SR 4.4.1.1 AND LCO 3.4.1.2/SR 4.4.1.2

This specification assures forced circulation of primary coolant. In POWER and LOW POWER, both loops with both steam generator sections (reheater and EES) are required to be operating and at least one helium circulator in each loop, depending on the power level, is required to be operating. Two-loop operation is required to protect the steam generator's internal components from overheating (FSAR Section 4.3.5.2). Single loop operation is permitted only long enough to reduce power in an orderly manner as required for recovery of the shutdown loop or to be below 5% power. (FSAR Section 4.3.4).

Specification 3/4.4.1.1 is applicable at power levels above 5% power. Below 5% power, single loop operation is acceptable. Also, as long as the CALCULATED BULK CORE TEMPERATURE remains below 760 degrees F (as determined per Specification 3.0.5), forced circulation is not required and may be interrupted as necessary.

At least two circulators (one in each loop) are required below 50% power, consistent with the operating guidelines provided in the circulator vendor's Operating and Maintenance Manual. Also, if both circulators in a loop TRIP, power is automatically reduced to about 50%. A single circulator TRIP results in a power reduction to about 50% power for conservatism. The requirements of Specification 3.4.1.1.b represent a power restriction, not a circulator restriction; i.e., there may be four operating circulators below 50% power.

The equipment that is used to provide forced circulation per this specification is also included in Specifications 3/4.5.1 and 3/4.5.3. The distinction between these specifications is that Specification 3.4.1 can be met using normal operating systems (normal feedwater to the EES sections, cold reheat to the reheaters, and steam to the circulator turbines), where Specifications 3.5.1 and 3.5.3 require SAFE SHUTDOWN COOLING capabilities (firewater through the emergency condensate and emergency feedwater headers to the EES sections, firewater through the emergency condensate header to the reheaters, and boosted firewater to the circulator water turbines).

Thirty minutes are allowed for a power reduction to below 50 percent if a circulator is lost, to allow the operators to confirm the automatic actions of the PLANT PROTECTIVE SYSTEM. See FSAR Section 7.1.2.6.

BASIS FOR SPECIFICATIONS LCO 3.4.1.1/SR 4.4.1.1 AND LCO 3.4.1.2/SR 4.4.1.2 (Continued)

Depressurization

In the unlikely event that all forced circulation is lost for 90 minutes, the PCRV is depressurized to reduce the density of the primary coolant and thereby reduce the heat transfer to the Liner Cooling System. This action is taken to maintain PCRV liner integrity, as discussed in FSAR Sections 14.10.3.1, D.1.1.1, and D.1.2.1.2. Start of depressurization is initiated as a function of prior power levels, with 2 hours from 100% RATED THERMAL POWER being the most limiting case. Operators will continue attempts to restore forced circulation cooling until 5 hours after the loss of forced circulation. (FSAR Section D.2.5). Multiple sources and flowpaths to establish forced convection cooling using circulators makes required depressurization highly unlikely. Cooldown using forced circulation cooldown is preferred to a depressurized cooldown with the PCRV liner cooling system, since forced cooling is required to assure prevention of fuel damage, depending on the plant power level. Depressurization of the PCRV under extended loss of forced circulation conditions is accomplished by venting the reactor helium through a train of the helium purification system and the reactor building vent stack filters to atmosphere. Start of depressurization times from various reactor power conditions are delineated in Figures 3.4.1-1, 3.4.1-2, and 3.4.1-3 and are discussed in the FSAR Section 9.4.3.3 and Appendix D.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of primary coolant helium flow.

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PRIMARY COOLANT

3/4.4.2 PRIMARY COOLANT ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. 2.40 curies-MeV/lb at 15 minutes after sampling (product of primary coolant noble gas beta plus gamma activity times E-BAR),

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

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

d. 140 curies per loop of bone dose equivalent Sr-90 plateout within the primary circuit.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: With the primary coolant circulating gaseous activity level exceeding Limit a or b above, or with the plateout activity level exceeding Limit c or d above, restore the primary coolant activity level to within the above limits within 48 hours or be in at least SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The primary coolant gross gaseous activity level shall be examined at least once per 24 hours by:

a. Use of the gross activity monitor (RT-9301), or

b. If the primary coolant gross activity monitor is inoperable, by collecting and analyzing a primary coolant sample.

4.4.2.2 The primary coolant gaseous and plateout activity levels shall be determined to be within the limits of Specification 3.4.2 as follows:

a. At least once per 7 days, by collecting and analyzing a grab sample of primary coolant. This grab sample analysis shall be used to determine the following:

1. E-BAR (See Note 1),

2. Curies - MeV/lb,

3. Plateout curies of DOSE EQUIVALENT I-131,

4. An estimate of the circulating DOSE EQUIVALENT I-131, and

5. An estimate of the total plateout Sr-90 activity level.

b. If the primary coolant activity level reaches 25% of the limits of Specification 3.4.2.a, b, or c above, at least once per 24 hours a grab sample of primary coolant shall be taken and analyzed per Specification 4.4.2.2.a above. Normal sample frequency (i.e., at least once per 7 days) may be resumed when the activity level is reduced to below 25% of the limits of Specification 3.4.2.a, b, or c, or when the activity level reaches a new equilibrium level, as defined by four consecutive daily samples whose results agree within 10% of the average of the four samples.

SPECIFICATION 4.4.2 (Continued)

c. One plateout probe shall be removed for evaluation coincident with the second, fourth, and sixth refueling, and at intervals not to exceed 5 REFUELING CYCLES thereafter. If, during the fifth REFUELING CYCLE, or any REFUELING CYCLE following the sixth REFUELING CYCLE, the primary coolant circulating gas activity is greater than 7,725 Ci, the plateout probe shall be removed at the end of that REFUELING CYCLE. The probes shall be analyzed for Sr-90 and I-131 inventory in the primary circuit. The results shall be used to determine the total plateout activity level of Sr-90 and the circulating activity of I-131 in the primary circuit.

4.4.2.3 The gross activity monitor (RT-9301) shall be demonstrated OPERABLE:

a. At least once per 31 days, by determining its sensitivity from the grab sample analysis from SR 4.4.2.2.

b. At least once per 18 months, by performance of a CHANNEL CALIBRATION.

NOTE 1: Calculations required to determine E-BAR shall consist of the following:

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

b. A determination of the average beta plus gamma energy per disintegration of each nuclide determined in NOTE 1.a above, by applying known decay energies and schemes, and

c. A calculation of E-BAR by appropriate weighting of each nuclide's beta and gamma energy with its concentration as determined in NOTE 1.a above.

BASIS FOR SPECIFICATION LCO 3.4.2/SR 4.4.2

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 dose to the skin of the whole body is a direct function of the gaseous beta activity in the primary coolant.

The primary coolant noble gas beta plus gamma concentration limit, (Specification 3.4.2.a) is based on the Maximum Credible Accident (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 reactor building exhaust system.

Correcting the noble gas 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 radionuclides during atmospheric transport to the EAB.

The U.S. Atomic Energy Commission Staff (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 SR-90 and I-131 Activity Limits

The equivalent Sr-90 and DOSE EQUIVALENT I-131 limits are based on the AEC's evaluation (Ref. 1) of 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 (Table 4.2 of Ref. 1) used a number of conservative assumptions to calculate the accident consequences. However, these assumptions result in calculated EAB doses which are well below 10 CFR 100 guidelines. The maximum equivalent activity levels (e.g., Sr-90 and I-131 limits) determined by the Commission staff from the Design Basis Accident No. 2 (DBA-2) are summarized in the following table:

BASIS FOR SPECIFICATION LCO 3.4.2/SR 4.4.2 (Continued)

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 doses calculated at the EAB, based on a dilution factor of 8.4 E-4 sec/m3 and dose conversion factors of 1,480 rem per milli-Curie of I-131 inhaled, and 36,700 rem per milli-Curie of Sr-90 inhaled (Ref. 3). More recent, less restrictive data, especially for Sr-90, is provided in Ref. 4. However, the AEC staff directed that the data from Ref. 3 be used.

Should information become available which leads to a change in the given dilution factors (e.g., Ref. 2), or should the data given in Ref. 4 become acceptable, a proposal for an amendment to the allowable activity levels (LCO limits) of this specification may be submitted.

Action Statement Bases

The ACTION statement permits 48 hours of continued operation to correct the situation. The 48 hour delay allows time for the activity level to decrease (due to decay, cleanup, and power reduction) and reach a new equilibrium level and is consistent with the Standard Technical Specifications for LWRs. If the situation cannot be corrected within 48 hours by cleanup, decay or power reduction then a reactor shutdown is appropriate.

BASIS FOR SPECIFICATION LCO 3.4.2/SR 4.4.2 (Continued)

BASIS for SR 4.4.2.1 and 4.4.2.2.a and b- Sampling and Analysis

Under normal operating conditions the gross activity of the primary coolant is measured and indicated on a continuous basis. The specified sampling interval provides an adequate check on the sensitivity of this monitoring equipment.

Once per 7 days, a grab sample is taken and analyzed to determine that the primary coolant activity levels are within their LCO limits. The circulating DOSE EQUIVALENT I-131 and the total plateout Sr-90 activity levels can only be determined from the plateout probe; these activity levels can be estimated from the grab sample analysis by calculations involving release rates and precursor measurements as discussed below.

If the gross activity monitor becomes inoperable, within 24 hours a sample will be taken and analyzed for its gross activity level. This will be continued on a 24 hour interval basis until the monitor is restored to OPERABLE status. The 24 hour sampling and analysis interval is adequate to ensure that 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 activity levels are calculated from grab samples and the readings of the gross activity 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 activity levels of any non-measured noble gas nuclides, Sr-90, and iodine nuclides necessary to compute DOSE EQUIVALENT I-131 are calculated by assuming that the release rate can be predicted from the release rate curves established using xenon and krypton noble gas nuclides.

BASIS FOR SPECIFICATION LCO 3.4.2/SR 4.4.2 (Continued)

BASIS for SR 4.4.2.2.c-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. e 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.

The 7,725 Ci in SR 4.4.2.2.c is 25% of the total of 30,900 Ci from the 842 Mw(t) "Design" column of FSAR Table 3.7-1, "Coolant Gas-Borne Activity."

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

$$A_{Sr-90}(t) = A_{Sr-90}(0)e^{-\lambda t} + \int_0^t A_{Kr-90}(\tau)e^{-\lambda(t-\tau)} \lambda \, d\tau$$

where $A_{Sr-90}(0)$ is the total Sr-90 activity level 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 weekly grab samples. Note that, if the Kr-90 activity is constant (or bounded, or can be averaged), the estimated Sr-90 activity level would be given by

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

BASIS FOR SPECIFICATION LCO 3.4.2/SR 4.4.2 (Continued)

This method of estimating the Sr-90 activity level 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 may be periodically updated by the plateout probe analyses to give the total measured Sr-90 plateout, regardless of origin.

The sensitivity of the gross activity monitor is calculated and posted once per 31 days. The sensitivity information is used by operators in the event of a radiological emergency. The frequencies for sensitivity check and calibration are equivalent to that for functional test and calibration of post accident coolant radiation monitors in the LWR STS.

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.

3. Report of Committee II on Permissible Dose for Internal Radiation, International Commission of Radiation Protection, Publ. 2, Health Physics 3, 1960.

4. Recommendations of the International Commission on Radiation Protection (as amended 1959 and revised 1962), ICRP Publ. Pergamon Press 1964.

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PRIMARY COOLANT

3/4.4.3 PRIMARY COOLANT IMPURITY LEVELS - HIGH TEMPERATURES

LIMITING CONDITION FOR OPERATION

3.4.3 a. The chemical impurity concentrations in the primary coolant shall not exceed 10 ppm (by volume) for the sum of H2O, CO, and CO2.

b. The chemical impurity concentrations in the primary coolant shall not exceed the above limit for more than 1000 ppm-days, integrated per REFUELING CYCLE.

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

ACTION: With chemical impurity concentrations (sum of H2O, CO, and CO2) exceeding:

a. 10 ppm but less than or equal to 100 ppm: reduce chemical impurity concentrations to 10 ppm or less within 10 days, or reduce CORE AVERAGE OUTLET TEMPERATURE to less than 1200 degrees F within the next 6 hours and comply with Specification 3.4.4.

b. 100 ppm but less than or equal to 1000 ppm: reduce chemical impurity concentration to below 100 ppm within 24 hours, or reduce CORE AVERAGE OUTLET TEMPERATURE to less than 1200 degrees F within the next 6 hours and comply with Specification 3.4.4.

c. 1000 ppm: reduce CORE AVERAGE OUTLET TEMPERATURE to less than 1200 degrees F within 1 hour, and comply with Specification 3.4.4

d. 1000 ppm-days integrated per REFUELING CYCLE: reduce chemical impurity concentration to 10 ppm or less or reduce CORE AVERAGE OUTLET TEMPERATURE to less than 1200 degrees F within 24 hours and comply with Specification 3.4.4.

SURVEILLANCE REQUIREMENTS

4.4.3.1 CHEMICAL IMPURITIES

The primary coolant shall be analyzed for H2O, CO, and CO2:

a. At least once per 24 hours when the sum of H2O, CO, and CO2 chemical impurity levels exceed 5 ppm, or

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

c. The integral of ppm-days (incremental concentrations above 10 ppm, for periods when the CORE AVERAGE OUTLET TEMPERATURE is greater than or equal to 1200 degrees F) in each REFUELING CYCLE shall be updated in conjunction with each analysis of the primary coolant impurities performed in accordance with a and b above.

4.4.3.2 PGX GRAPHITE SURVEILLANCE

The PGX graphite surveillance specimens (installed in bottom transition reflector elements as indicated in Table 4.4.3.2-1) shall be removed at refueling intervals shown in Table 4.4.3.2-1 unless the progressive examination of the specimens dictates 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 prior to the next refueling outage and shall also be utilized to modify, as necessary, the planned removal of subsequent PGX surveillance specimens.

The results of these examinations and assessments shall be submitted to the Commssion staff for review prior to the next refueling outage.

4.4.3.3 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, during the refueling outages specified in Table 4.4.3.2-1, when the PGX graphite specimens are removed from the core in accordance with SR 4.4.3.2.

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PGX GRAPHITE SURVEILLANCE

TABLE 4.4.3.2-1

TRANSITION 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

*The 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.

BASIS FOR SPECIFICATION LCO 3.4.3/SR 4.4.3

For plant operation in the range of about 25% to 100% RATED THERMAL POWER, maximum impurity levels have been established to restrict graphite oxidation and carbon transport from the reactor core to cooler portions of the primary coolant system to about 330 lb/yr. Limiting the quantity of graphite oxidized or carbon transported from the reactor core ensures the structural integrity of the fuel elements and the core support structure, and limits the carbon deposition 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 which is within allowances assumed in the design. (FSAR 4.2.1).

Primary coolant is monitored and alarmed by the PPS Dewpoint Moisture Monitoring System. The Primary Coolant Pressure-High instrumentation would also indicate the presence of impurities in the Primary Coolant System.

PGX graphite specimens have been placed in modified coolant channels in five transition reflector elements in the hottest columns of regions 22, 24, 25, 27, and 30. The surveillance test specimens are subjected to the same primary coolant conditions, as well as other reactor parameters, as seen by the PGX core support blocks. Examination and tests of the surveillance test specimens at regular intervals can readily be utilized to assess oxidation rates, oxidation profiles, as well as general degradation of the PGX core support blocks to predict adequately the structural integrity of the core support blocks over the operating life of the reactor. (FSAR Section 3.3.2.2 and Appendix A.12.5.5).

Visual examination of the core support blocks in those regions chosen for insertion of PGX graphite specimens provides 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. Consequently, any cracking would be expected to originate at these locations, and should be discovered during inspection.

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BASIS FOR SPECIFICATION LCO 3.4.3/SR 4.4.3 (Continued)

The PGX graphite surveillance specimen examination, evaluation, assessment and reporting prior to the next refueling outage provide ample time to determine if the next specimens should be withdrawn at an earlier refueling number than shown in Table 4.4.3.2-1.

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PRIMARY COOLANT

3/4.4.4 PRIMARY COOLANT IMPURITY LEVELS - LOW TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

a.

IMPURITY	LIMIT
H2O	The dew point limits shown in Figure 3.4.4-1 as "Acceptable" or "Limited Acceptable".
CO2	1,000 ppm (by volume)
CO	15,000 ppm (by volume)

b. The cumulative time during which the dew point is in the "Limited Acceptable" region on Figure 3.4.4-1 shall not exceed a total of 90 days during any one REFUELING CYCLE.

APPLICABILITY: POWER, LOW POWER, and STARTUP, when CORE AVERAGE OUTLET TEMPERATURE is less than 1200 degrees F.

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SPECIFICATION 3.4.4 (Continued)

ACTION:

a. With primary coolant H2O impurity concentration exceeding the specified limit, reduce the concentration to within "Acceptable" or "Limited Acceptable" values per Figure 3.4.4-1 within 1 hour or reduce CORE AVERAGE OUTLET TEMPERATURE to less than 725 degrees F within the next 6 hours. If concentrations cannot be restored to within the specified limits within this 6 hours be in SHUTDOWN within the following 6 hours.

b. With primary coolant moisture dew point in the "Limited Acceptable" region of Figure 3.4.4-1 for greater than 90 cummulative days during a REFUELING CYCLE, discontinue operation in the "Limited Acceptable" region for the remainder of the REFUELING CYCLE.

c. With primary coolant CO2 or CO impurity concentration exceeding the specified limit, reduce the concentration to within the specified limit within 6 hours or be in SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The primary coolant shall be analyzed for H2O, CO, and CO2:

a. At least once per 24 hours when the chemical impurity concentrations exceed 50% of any of the above limits, or

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

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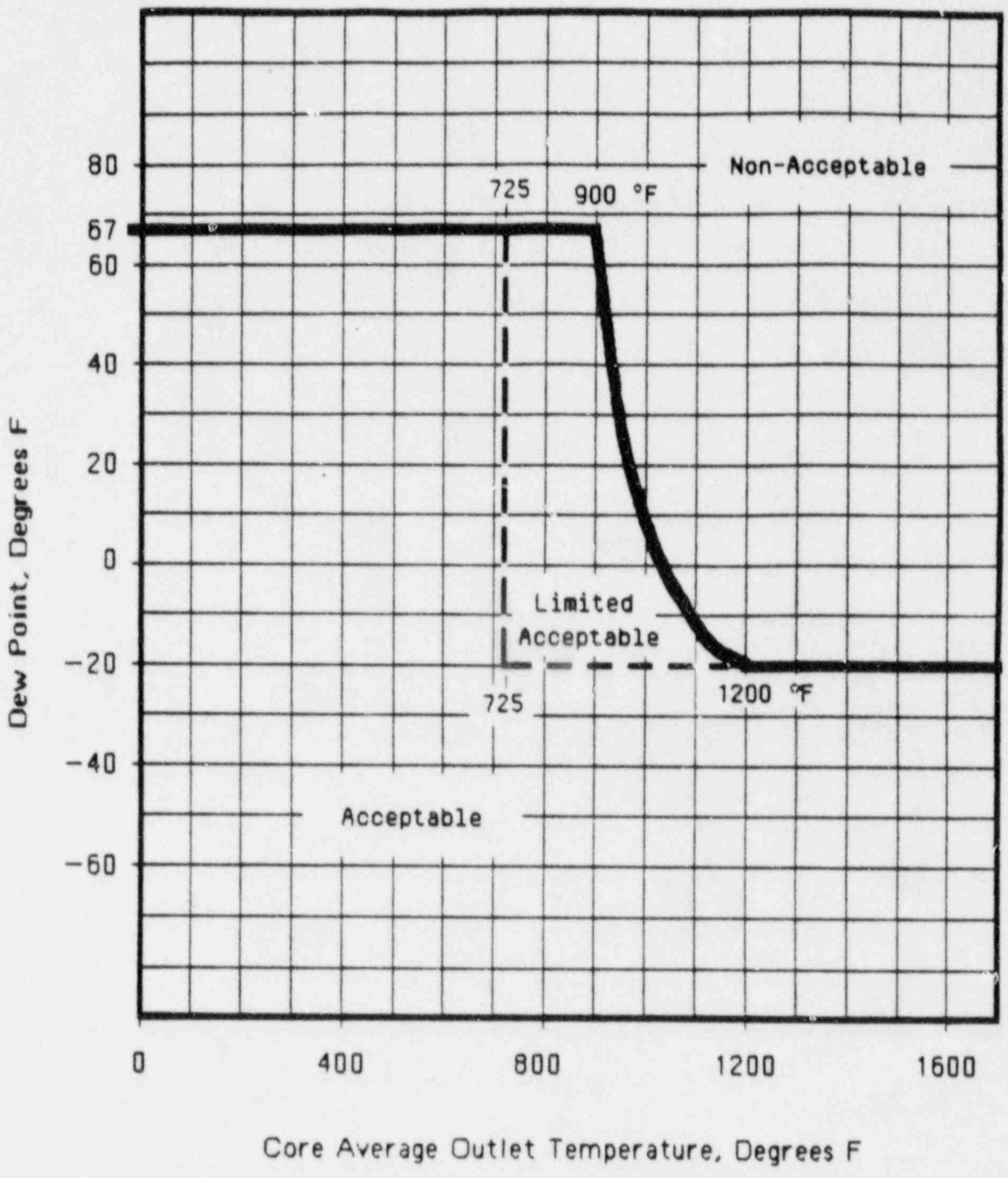

PRIMARY COOLANT MOISTURE LEVELS
(Applicable During POWER, LOW POWER, and STARTUP)

Figure 3.4.4-1

BASIS FOR SPECIFICATION LCO 3.4.4/SR 4.4.4

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

There is a need to prevent corrosion of metals in the primary coolant system and to 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 (B2O3). 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:

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

b) 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 B4C oxidation for allowable impurity levels (Specification 3.4.4).

The criterion used to establish the limits of Figure 3.4.4-1 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 less than the minimum core SHUTDOWN MARGIN of 0.01 delta k (Specification 3.1.4), and only about one-half of the reactivity anomaly of 0.01 delta k specified in Specification 3.1.7.

BASIS FOR SPECIFICATION LCO 3.4.4/SR 4.4.4 (Continued)

The requirement used in developing the curve of Figure 3.4.4-1 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 requirement with the dew point limits shown in Figure 3.4.4-1 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 90-day ACTION statement is conservative, since operation is limited to 90 days per REFUELING CYCLE only on the dew point curve itself. Operation in excess of 90 days per REFUELING CYCLE could be allowed in the "Limited Acceptable" regime without violating the criteria which serve as the BASIS of LCO 3.4.4.

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 4.4.4 will ensure that chemical impurities will be measured and controlled to minimize corrosion of core materials.

Primary coolant is monitored and alarmed by the Analytical Moisture Monitoring System and the PPS Dewpoint Moisture Monitoring System. The Primary Coolant Pressure-High instrumentation would also indicate the presence of impurities in the Primary Coolant System.

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.1 HELIUM CIRCULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 At least one helium circulator in each primary coolant loop shall be OPERABLE including its water turbine.

APPLICABILITY: POWER, LOW POWER, STARTUP*, SHUTDOWN*, and REFUELING*

ACTION: a. With no OPERABLE helium circulator in one of the two primary coolant loops, restore the inoperable equipment to OPERABLE status within 72 hours, or

1. When in POWER, LOW POWER, or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

2. When in SHUTDOWN or REFUELING, suspend all operations involving CORE ALTERATIONS or control rod movement resulting in positive reactivity changes.

b. With no OPERABLE helium circulator in either primary coolant loop, restore one helium circulator to OPERABLE status within 1 hour, or

1. When in POWER, LOW POWER, or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

2. When in SHUTDOWN or REFUELING, suspend all operations involving CORE ALTERATIONS or control rod movement resulting in positive reactivity changes.

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* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

SURVEILLANCE REQUIREMENTS

4.5.1.1 The helium circulators shall be demonstrated OPERABLE:

a. At least once per REFUELING CYCLE whereby circulators 1B and 1D will be tested during even numbered cycles and circulators 1A and 1C during odd numbered cycles, by demonstrating operation on water turbine drive of each circulator and verifying equivalent 3.8% rated helium flow on condensate at reduced pressure (to simulate firewater pump discharge) using each emergency water booster pump (P-2109 and P-2110) and the emergency condensate header.

b. At least once per REFUELING CYCLE by monitoring the proper closure of the helium shutoff valves.

c. At least once per 10 years by verifying:

1. A helium circulator compressor wheel rotor, turbine wheel and Pelton wheel are free of 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. Testing shall be scheduled so that over 4 inspection periods, each circulator will be tested once. Other helium circulator components, accessible without further disassembly than required to inspect these wheels, shall be visually examined, and

2. At least 10% of primary coolant pressure boundary bolting and other structural bolting has been removed for the inspection above, and it is free of inherent or developed defects.

3. Reports

Within 90 days of examination completion, a Special Report shall be submitted to the Commission in accordance with Specification 6.9.2. This report shall include the results of the helium circulator examinations.

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SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.1 HELIUM CIRCULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.2 At least one helium circulator shall be OPERABLE including its water turbine.

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING*

ACTION: With no OPERABLE helium circulator, be in at least SHUTDOWN within 12 hours and restore the required equipment to OPERABLE status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, or suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.5.1.2 The helium circulators shall be demonstrated OPERABLE by the performance of SR 4.5.1.1.

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* Whenever CALCULATED BULK CORE TEMPERATURE is less than or equal to 760 degrees F.

BASIS FOR SPECIFICATIONS LCO 3.5.1.1/SR 4.5.1.1 AND LCO 3.5.1.2/SR 4.5.1.2

The scope of this specification includes the helium circulator machine, with particular emphasis on the water turbine (Pelton wheel) drive, and on the water supply piping out to the speed control valves. The connecting supply piping and turbine water drain piping is included in Specification 3/4.5.2.

One helium circulator (and one steam generator EES section) ensures SAFE SHUTDOWN COOLING when the plant is pressurized. One helium circulator in each primary coolant loop is specified during POWER, LOW POWER, STARTUP, SHUTDOWN, and REFUELING with CALCULATED BULK CORE TEMPERATURE greater than 760 degrees F to allow for a single failure. One circulator, operating with motive power from either (a) condensate or boosted firewater supplied via the emergency condensate header, or (b) feedwater or boosted firewater supplied via the emergency feedwater header, provides sufficient primary coolant circulation for the pressurized condition. SAFE SHUTDOWN COOLING is discussed in FSAR Section 10.3.9, single failure considerations in Section 10.3.10, and condensate and boosted firewater cooldown transients in FSAR Sections 14.4.2.1 and 14.4.2.2.

The helium circulators, with feedwater supplied to their water turbines, provide sufficient cooling in the event of a PCRV depressurization accident. Feedwater supply is assured by Specification 3/4.7.1.1, as this is not SAFE SHUTDOWN COOLING equipment. Two circulators, operating with emergency water drive, supplied with feedwater via the emergency feedwater header, provide sufficient primary coolant circulation following a postulated Design Basis Depressurization Accident (DBA-2). (FSAR Section 14.11.2). DBA-2 is a highly incredible event the probability of which has been determined to be approximately 1.0E-7 per year, and for which protection against single failures is not a feature of FSV (FSAR Section 14.11.1). For the Maximum Credible Depressurization Accident (MCA), a single helium circulator with feedwater drive provides sufficient circulation, as discussed in FSAR Section 14.4.3.2.

The SAFE SHUTDOWN COOLING emergency water drive source is boosted firewater, which is included in Specification 3/4.5.5. The helium circulator Pelton wheels can be driven by condensate from the condensate pumps (either 60% or 12 1/2%) or by feedwater from the boiler feedwater pumps; however, these are not SAFE SHUTDOWN COOLING equipment.

BASIS FOR SPECIFICATIONS LCO 3.5.1.1/SR 4.5.1.1 and LCO 3.5.1.2/SR 4.5.1.2 (Continued)

Redundancy Criteria

The use of 760 degrees F CALCULATED BULK CORE TEMPERATURE as a division between the APPLICABILITY of Specification 3.5.1.1 verses 3.5.1.2 is explained in the BASIS for Specification 3.0.5.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of primary coolant helium flow.

ACTIONS

With CALCULATED BULK CORE TEMPERATURE above 760 degrees F, an inoperable helium circulator is permitted for 72 hours, as this constitutes a loss of redundancy in SAFE SHUTDOWN COOLING equipment. 72 hours is consistent with Actions required in Light Water Reactors for loss of redundancy in Emergency Core Cooling System equipment. If the inoperable helium circulator cannot be repaired within this time, an orderly shutdown is required.

With the CALCULATED BULK CORE TEMPERATURE below 760 degrees F, 12 hours are allowed to be in at least SHUTDOWN, as this is a reasonable time period for an orderly shutdown from STARTUP. Then, as long as the inoperable equipment is restored to OPERABLE status prior to reaching 760 degrees F, no further Actions are required.

OPERABILITY Demonstration

Each helium circulator is tested for its SAFE SHUTDOWN COOLING capabilities every other REFUELING CYCLE. This is done by simulating firewater flow by throttling condensate flow, boosting it with the emergency water booster pumps, and verifying that PRIMARY COOLANT FLOW rate is equivalent to 3.8% at full density with cold core conditions. Cold core conditions result in reduced helium bouyancy, and thus, higher developed flow rates.

A 10-year ISI inspection is required to ensure helium circulator integrity.

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.2 HELIUM CIRCULATOR AUXILIARIES

LIMITING CONDITION FOR OPERATION

3.5.2.1 At least the following helium circulator water turbine drive capabilities and auxiliary equipment shall be OPERABLE for the helium circulators that are required to be OPERABLE:

a. Two SAFE SHUTDOWN COOLING (firewater) supplies for each water turbine drive. This includes two emergency water booster pumps (P-2109 and P-2110) and a flow path from both the emergency feedwater and emergency condensate headers to and including the helium circulator water turbine speed control valves,

b. The turbine water removal system, including two turbine water removal pumps (P-2103 and P-2103S),

c. The normal bearing water system, including two normal bearing water pumps per loop (P-2101, P-2101S, P-2106 for Loop 1; P-2102, P-2102S, P-2107 for Loop 2), and two bearing water makeup pumps (P-2105 and P-2108) with two supply sources of bearing water makeup,

d. The associated bearing water accumulator (T-2114 with gas pressurizer T-2112 for Loop 1; T-2115 with gas pressurizer T-2113 for Loop 2), and

e. The supply and discharge valve interlocks ensuring automatic water turbine start capability.#

APPLICABILITY: POWER, LOW POWER, STARTUP*, SHUTDOWN*, and REFUELING*

ACTION:

a. With any one of the above required SAFE SHUTDOWN COOLING water turbine drive supplies, turbine water removal pumps, bearing water pumps per loop, bearing water makeup pumps, or sources of bearing water makeup inoperable, restore the inoperable equipment to OPERABLE status within 72 hours, or be in at least SHUTDOWN within the next 24 hours.

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* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

\# The supply and discharge valve interlocks ensuring automatic water turbine start capability are only required to be OPERABLE in POWER.

SPECIFICATION LCO 3.5.2.1 (Continued)

b. With less than the above required OPERABLE bearing water accumulators, restore the inoperable equipment to OPERABLE status within 24 hours, or be in at least SHUTDOWN within the next 24 hours.

c. With less than the above required OPERABLE valve interlocks, restore the inoperable equipment to OPERABLE status within 72 hours, or be in at least LOW POWER within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.2.1 The helium circulator auxiliaries shall be demonstrated OPERABLE:

a. At least once per 92 days by:

1. Testing the bearing water accumulators and their actuation valves,

2. Performing a turbine water removal pump start test based on a simulated drain tank level to verify automatic actuation and pump start capability, and

3. Performing a start test of bearing water makeup pump P-2105, based on a simulated low pressure in the backup bearing water supply line to verify automatic actuation and pump start capability, and operating P-2105 in the recycle mode.

4. Performing a functional test of the emergency bearing water makeup pump (P-2108).

b. At least once per REFUELING CYCLE by:

1. Testing the water turbine inlet and outlet valve interlocks ensuring automatic water turbine start capability by simulating a PPS signal resulting from one loop being tripped and the circulators' steam turbine drives in the operating loop having been tripped.

2. Performing a functional test of each emergency water booster pump. This is performed in conjunction with Specification SR 4.5.1.1.a.

c. At each scheduled plant shutdown by functionally testing each bearing water pump.

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.2 HELIUM CIRCULATOR AUXILIARIES

LIMITING CONDITION FOR OPERATION

3.5.2.2 At least the following helium circulator water turbine drive capabilities and auxiliary equipment shall be OPERABLE for each helium circulator that is required to be OPERABLE:

a. One SAFE SHUTDOWN COOLING (firewater) supply for each water turbine drive. This includes one emergency water booster pump (P-2109 or P-2110), and a flow path from either the emergency feedwater or the emergency condensate header to and including the helium circulator water turbine speed control valve.

b. The turbine water removal system, including one turbine water removal pump (P-2103 or P-2103S),

c. The normal bearing water system, including two bearing water pumps (P-2101, P-2101S, P-2106 for Loop 1; P-2102, P-2102S, P-2107 for Loop 2), and one bearing water makeup pump (P-2105 or P-2108), with one supply source of bearing water makeup.

d. The associated bearing water accumulator (T-2114 with gas pressurizer, T-2112 for Loop 1; T-2115 with gas pressurizer, T-2113 for Loop 2).

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING*

ACTION: With less than the above required OPERABLE equipment, be in at least SHUTDOWN within 12 hours, and restore the required equipment to OPERABLE status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, or suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.5.2.2 The helium circulator auxiliaries shall be demonstrated OPERABLE by performance of SR 4.5.2.1.

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* Whenever CALCULATED BULK CORE TEMPERATURE is less than or equal to 760 degrees F.

BASIS FOR SPECIFICATIONS LCO 3.5.2.1/SR 4.5.2.1 AND LCO 3.5.2.2/SR 4.5.2.2

The scope of this specification includes the helium circulator Pelton wheel supply piping from the emergency feedwater or emergency condensate headers (which are specifically included in Specification 3/4.5.4) to and including the helium circulator water turbine speed control valves. The helium circulator Pelton wheel drives are included in Specification 3/4.5.1.

Whenever the CALCULATED BULK CORE TEMPERATURE (CBCT) is greater than 760 degrees F, two SAFE SHUTDOWN COOLING supplies are specified to ensure forced circulation in the event of a single active failure in this piping.

One turbine water removal pump has sufficient capacity to remove the water from two circulator water turbines. Also, the turbine water removal tank overflow to the reactor building sump is available if the normal pump flow path is lost.

Each of the two separate and independent recirculating bearing water loops supplies the bearing water requirements of the two helium circulators in a primary coolant loop. Each of the normal bearing water loops contains three bearing water pumps in series. Each pump develops full flow and one-half the required head rise. Two pumps are running, the third is a standby. Requiring two pumps to be OPERABLE when the CBCT is greater than 760 degrees F is acceptable because loss of a second bearing water pump in a loop could result in a loop shutdown but it would not affect the plant's ability to shut down on the other loop.

Makeup bearing water requirements are normally obtained from the feedwater system. A separate bearing water makeup pump is provided as a backup to supply makeup water to the bearing water surge tank. The bearing water makeup pump (P-2105) takes suction from either the deaerator, the condensate storage tanks (normal), or the firewater system (emergency). If this pump is inoperative, an emergency bearing water makeup pump (P-2108) can supply water at a reduced capacity from the condensate storage tank, or the firewater system (emergency).

A 72-hour ACTION time for all the above equipment is appropriate whenever the CBCT is greater than 760 degrees F as this is a loss of redundancy in SAFE SHUTDOWN COOLING equipment. 72 hours is consistent with actions required in Light Water Reactors for loss of redundancy in Emergency Core Cooling System equipment.

BASIS FOR SPECIFICATIONS LCO 3.5.2.1/SR 4.5.2.1 and LCO 3.5.2.2/SR 4.5.2.2 (Continued)

When the CALCULATED BULK CORE TEMPERATURE is less than 760 degrees F, the ACTION to be shut down within 12 hours allows for an orderly shutdown from STARTUP.

Each bearing water loop contains a gas pressurizer and bearing water accumulator capable of supplying bearing water for 30 seconds at design flow rate if no other source of bearing water is available. This is adequate for shutdown of the affected circulators without damage to the bearings.

The bearing water system, including the bearing water accumulators and the bearing water makeup pumps, is functionally tested at 31-day and 92-day intervals to ensure proper operation. There is no redundancy in the bearing water accumulators, and a 24 hour ACTION statement restoration time is provided. This is acceptable considering the low likelihood of failures of these components.

Auto water turbine start is prevented if a water turbine TRIP exists or the auto water turbine start control switch is not in the auto position. The aforementioned interlock circuitry is tested once per REFUELING CYCLE, to ensure proper system operation. The automatic water turbine start feature is relied upon in the event the control room has to be abandoned. Since this is an unlikely event, a 72-hour ACTION statement restoration time is acceptable. Also, the ACTION to reduce power to the LOW POWER mode is appropriate because the interlocks are only required in the POWER mode.

Redundancy Criteria

The use of 760 degrees F CALCULATED BULK CORE TEMPERATURE as a division between the APPLICABILITY of Specification 3.5.2.1 versus 3.5.2.2 is explained in the BASIS for Specification 3.0.5.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of primary coolant helium flow.

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.3 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.5.3.1 The steam generator in each primary coolant loop with an OPERABLE helium circulator shall be OPERABLE, including both the economizer-evaporator-superheater (EES) and reheater sections.

APPLICABILITY: POWER, LOW POWER, STARTUP*, SHUTDOWN*, and REFUELING*

ACTION:

a. With one of the required steam generator sections inoperable, restore the inoperable equipment to OPERABLE status within 72 hours, or

   1. When in POWER, LOW POWER, or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

   2. When in SHUTDOWN or REFUELING, suspend all operations involving CORE ALTERATIONS or control rod movement resulting in positive reactivity changes.

b. With any two of the required steam generator sections inoperable, restore at least one section to OPERABLE status within 1 hour, or

   1. When in POWER, LOW POWER, or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

   2. When in SHUTDOWN or REFUELING, suspend all operations involving CORE ALTERATIONS or control rod movement resulting in positive reactivity changes.

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* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The steam generators shall be demonstrated OPERABLE:

a. At least once per 18 months by:

1. Verifying proper flow through the emergency feedwater header and emergency condensate header to the EES sections, and verifying proper flow through the emergency condensate header to the reheater sections.

2. Verifying the outlet flow path by cycling the valves in the six inch vent lines and observing that the vent flowpaths are not obstructed.

b. At least once per 5 years by volumetrically examining the accessible portions of the following bimetallic welds for indications of subsurface defects:

1. The main steam ring header collector to main steam piping weld for one steam generator module in each loop, and

2. The main steam ring header collector to collector drain piping weld for one steam generator module in each loop, and

3. 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 a steam generator EES tube is plugged due to a leak, specimens from the accessible subheader tubes connected to the leaking inaccessible tubes shall be metallographically examined. Reheater subheater tubes are not accessible.

The results of this metallographic examination shall be compared to the results from the specimens of all previous tube leaks.

SPECIFICATION SR 4.5.3.1 (Continued)

A study shall be performed to evaluate the size and elevation of the 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 SR 4.5.3.1.b,

   b) Continued operation considering the condition of the steam generator materials, and

   c) OPERABILITY of the steam generator sections considering the number of plugged tubes and their ability to remove decay heat.

2. Reports

   Within 90 days of the return to operation following each steam generator tube leak study 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), at least the preliminary results of the metallographic and engineering analyses performed, the postulated cause of the leak if identified, and corrective action to be taken.

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.3 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.5.3.2 At least one steam generator in one primary coolant loop with an OPERABLE helium circulator shall be OPERABLE, including at least the economizer-evaporator-superheater (EES) or the reheater section.

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING*

ACTION: With no OPERABLE steam generator section, be in at least SHUTDOWN within 12 hours and restore the inoperable equipment to OPERABLE status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, or suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.5.3.2 The steam generators shall be demonstrated OPERABLE by the performance of SR 4.5.3.1.

* Whenever CALCULATED BULK CORE TEMPERATURE is less than or equal to 760 degrees F.

BASIS FOR SPECIFICATIONS LCO 3.5.3.1/SR 4.5.3.1 AND LCO 3.5.3.2/SR 4.5.3.2

The requirements for OPERABLE steam generator(s) provide an adequate means for removing heat from the primary coolant system to the secondary coolant system. The helium flow which cools the reactor core enters the steam generator at high temperature and gives up its heat to the reheat steam section and the EES section.

The steam generator sections are essentially passive heat exchangers and are OPERABLE when their pressure boundaries are intact and when they are capable of receiving flow from their respective emergency feedwater or condensate headers. This is separate from their normal feedwater flow path.

Each steam generator consists of six identical individual steam generator modules operating in parallel. Each module consists of a reheater section and an EES section. Any one EES section provides sufficient heat removal capability to ensure SAFE SHUTDOWN COOLING. The reheater sections are also used in certain accident analyses, such as wrong loop dump with an EES leak, (FSAR Section 14.5.3.2), but they are not relied upon for SAFE SHUTDOWN COOLING.

During POWER, LOW POWER, STARTUP, SHUTDOWN, and REFUELING with CALCULATED BULK CORE TEMPERATURE greater than 760 degrees F, both steam generator sections in both loops are required to be OPERABLE. This allows for a single failure and ensures an OPERABLE EES section for SAFE SHUTDOWN COOLING.

During SHUTDOWN and REFUELING with CALCULATED BULK CORE TEMPERATURE less than or equal to 760 degrees F, redundancy is not required and either the reheater section or the EES section of one steam generator can be used for shutdown heat removal from the primary coolant.

Redundancy Criteria

The use of 760 degrees F CALCULATED BULK CORE TEMPERATURE as a division between the APPLICABILITY of Specification 3.5.3.1 versus 3.5.3.2 is explained in the BASIS for Specification 3.0.5.

BASIS FOR SPECIFICATIONS LCO 3.5.3.1/SR 4.5.3.1 AND LCO 3.5.3.2/SR 4.5.3.2 (Continued)

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, or primary coolant helium flow.

ACTIONS

With CALCULATED BULK CORE TEMPERATURE above 760 degrees F, an inoperable steam generator section is permitted for 72 hours, as this constitutes a loss of redundancy in SAFE SHUTDOWN COOLING equipment. 72 hours is consistent with Actions required in Light Water Reactors for loss of redundancy in Emergency Core Cooling System equipment. If the inoperable section cannot be repaired within this time, an orderly shutdown is required.

With the CALCULATED BULK CORE TEMPERATURE below 760 degrees F, 12 hours are allowed to be in at least SHUTDOWN, consistent with the time allowed for orderly shutdown from STARTUP. Then, as long as the inoperable equipment is restored to OPERABLE status prior to reaching 760 degrees F, no further Actions are required.

OPERABILITY Demonstrations

The steam generator EES sections can receive water from either the associated emergency condensate header or the emergency feedwater header. Also, the reheater sections can receive water from the emergency condensate header.

During shutdown conditions, the required flowpaths to all steam generator sections are demonstrated. Also, the six inch vent valves are cycled and the outlets are visually examined for obstructions; actual flow is not demonstrated.

A six-inch vent line to atmosphere from each EES discharge header is included in the SAFE SHUTDOWN COOLING flowpath to reduce downstream flow resistance and allow higher flowrates. (FSAR Section 10.3.9).

BASIS FOR SPECIFICATIONS LCO 3.5.3.1/SR 4.5.3.1 AND LCO 3.5.3.2/SR 4.5.3.2 (Continued)

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 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.

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 approximate 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.

BASIS FOR SPECIFICATIONS LCO 3.5.3.1/SR 4.5.3.1 AND LCO 3.5.3.2/SR 4.5.3.2 (Continued)

Because of the subheader designs leading to the steam generator tube bundles, 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 surface of the subheader, giving an indication of the conditions of the leaking tube internal surface, 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 ensure that the consequences of postulated tube leaks remain within the limits analyzed in the FSAR.

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.4 EMERGENCY CONDENSATE AND EMERGENCY FEEDWATER HEADERS

EMERGENCY CONDENSATE AND EMERGENCY FEEDWATER HEADERS - OPERATION

LIMITING CONDITION FOR OPERATION

3.5.4.1 The emergency condensate header and the emergency feedwater header shall be OPERABLE.

APPLICABILITY: POWER, LOW POWER, STARTUP*, SHUTDOWN*, and REFUELING*

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:

   1. When in POWER, LOW POWER, or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

   2. When in SHUTDOWN or REFUELING, suspend all operations involving CORE ALTERATIONS or control rod movement resulting in positive reactivity changes.

b. With both the emergency condensate header and the emergency feedwater header inoperable, restore one header to OPERABLE status within 1 hour, or

   1. When in POWER, LOW POWER, or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

   2. When in SHUTDOWN or REFUELING, suspend all operations involving CORE ALTERATIONS or control rod movement resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.5.4.1 No additional Surveillance Requirements are required other than those surveillances identified in Specifications SR 4.5.1.1.a, SR 4.7.1.1, and SR 4.5.3.1.a.1.

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* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.4 EMERGENCY CONDENSATE AND EMERGENCY FEEDWATER HEADERS

EMERGENCY CONDENSATE AND EMERGENCY FEEDWATER HEADERS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.4.2 Either the emergency condensate header or the emergency feedwater header shall be OPERABLE.

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING*

ACTION: With both the emergency feedwater and emergency condensate header inoperable, be in at least SHUTDOWN within 12 hours and restore at least one header to OPERABLE status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F or suspend all operations involving CORE ALTERATIONS or control rod movement resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.5.4.2 No additional Surveillance Requirements are required other than those surveillances identified in Specification SR 4.5.4.1.

* Whenever CALCULATED BULK CORE TEMPERATURE is less than or equal to 760 degrees F.

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BASIS FOR SPECIFICATION LCO 3.5.4/SR 4.5.4

The OPERABILITY of the emergency condensate header and the emergency feedwater header ensures redundant water supply paths to the helium circulators and steam generators for SAFE SHUTDOWN COOLING 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 cooling capability. OPERABILITY of the aforementioned headers is accomplished during SHUTDOWN by verifying flow through each header to the steam generators and helium circulators.

Both the emergency condensate header and the emergency feedwater header are considered SAFE SHUTDOWN COOLING equipment.

Redundancy Criteria

The use of 760 degrees F CALCULATED BULK CORE TEMPERATURE as a division between the APPLICABILITY of Specification 3.5.4.1 versus 3.5.4.2 is explained in the BASIS for Specification 3.0.5.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of primary coolant helium flow.

The emergency feedwater header is not normally placed in service until approximately 30% reactor power, to prevent unnecessary long-term wear of components associated with the emergency feedwater header. Nevertheless it is still required to be OPERABLE during the aforementioned MODES.

BASIS FOR SPECIFICATION LCO 3.5.4/SR 4.5.4 (Continued)

ACTIONS

With CALCULATED BULK CORE TEMPERATURE above 760 degrees F, an inoperable header is permitted for 72 hours, as this constitutes a loss of redundancy in SAFE SHUTDOWN COOLING equipment. 72 hours is consistent with actions required in Light Water Reactors for loss of redundancy in Emergency Core Cooling System equipment. If the inoperable header cannot be repaired within this time, an orderly shutdown is required.

With the CALCULATED BULK CORE TEMPERATURE below 760 degrees F, 12 hours are allowed to be in at least SHUTDOWN, consistent with the time allowed for an orderly shutdown from STARTUP. Then, as long as the inoperable equipment is restored to OPERABLE status prior to reaching 760 degrees F, no further actions are required.

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SAFE SHUTDOWN COOLING SYSTEMS

3/4.5.5 SAFE SHUTDOWN COOLING WATER SUPPLY SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.5 The SAFE SHUTDOWN COOLING water supply system shall be OPERABLE with:

a. Two circulating water makeup storage ponds OPERABLE with a minimum combined inventory of 20 million gallons,

b. At least two OPERABLE circulating water makeup pumps (P-4118-P, P-4118S-P, or P-4118SX-P) connectible to an essential bus,

c. Two OPERABLE firewater pumps, both the motor driven (P-4501) and the engine driven (P-4501S), including the associated pump pits and at least 370 gallons of fuel in the day tank for the engine driven pump, and

d. Two OPERABLE flow paths capable of taking suction from the circulating water makeup storage ponds and transferring the water via the circulating water makeup pumps and the firewater pumps to the firewater header that supplies SAFE SHUTDOWN COOLING equipment.

APPLICABILITY: At all times

ACTION:

a. With CALCULATED BULK CORE TEMPERATURE greater than 760 degrees F:

1. With any single component required by Specification 3.5.5.a, b, or c above inoperable, restore the inoperable equipment to OPERABLE status within 72 hours or be in at least SHUTDOWN within the next 24 hours.

2. With more than one component required by Specification 3.5.5.a, b, or c above inoperable, but with one OPERABLE flowpath through any combination of the above required components, restore the inoperable equipment to OPERABLE status within 48 hours or be in at least SHUTDOWN within the next 24 hours.

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SPECIFICATION LCO 3.5.5 (Continued)

3. With no OPERABLE flowpath through the above required components, restore at least one flowpath to OPERABLE status within 1 hour, or

   a) When in POWER, LOW POWER, or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

   b) When in SHUTDOWN or REFUELING, suspend all operations involving CORE ALTERATIONS or control rod movement resulting in positive reactivity changes.

b. With CALCULATED BULK CORE TEMPERATURE less than or equal to 760 degrees F:

   1. With only one OPERABLE flowpath through any combination of the above required components, restore the inoperable equipment to OPERABLE status within 14 days or provide alternate backup equipment or water supply. The provisions of Specification 3.0.4 are not applicable.

   2. With no OPERABLE flowpath through the above required components, be in at least SHUTDOWN within 12 hours and restore at least one flowpath to OPERABLE status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, or suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.5.5.1 The SAFE SHUTDOWN COOLING water supply system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume in each of the circulating water makeup ponds.

b. At least once per 31 days by:

   1. Starting the electric motor-driven firewater pump and operating it for at least 15 minutes,

   2. Starting each circulating water makeup pump that is not already running, and

   3. Verifying that each valve in the flow path, that is not locked, sealed, or otherwise secured in place is in its correct position.

SPECIFICATION SR 4.5.5.1 (Continued)

c. At least once per 12 months by:

1. Cycling each valve in the flow path that is testable during plant operation, through at least one complete cycle of full travel, and

2. Verifying the performance capability and mechanical condition of each circulating water makeup pump. This may be performed at the next scheduled plant shutdown, if not performed during the previous 12 months.

d. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that the automatic valve in the flow path actuates to its correct position,

2. Verifying that each firewater pump (motor-driven and engine-driven) develops at least 1425 gpm at 119 psig,

3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4. Verifying that each firewater pump starts (sequentially) to maintain system pressure greater than or equal to 119 psig.

e. At least once per 5 years by verifying the alignment and settlement of the circulating water makeup pond embankments, and by examining the embankments and water structures for abnormal erosion, cracks, seepage, leakage, and accumulation of silt or debris.

4.5.5.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. The fuel day tank contains at least 370 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel day tank, obtained in accordance with ASTM-D270-1975 is within the acceptable limits specified in Table 1 of ASTM-D975-1977 when checked for viscosity, water and sediment.

SPECIFICATIONS SR 4.5.5.2 (Continued)

c. At least once per 18 months, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction its manufacturer's recommendations for the class of service.

4.5.5.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

BASIS FOR SPECIFICATION LCO 3.5.5/SR 4.5.5

The SAFE SHUTDOWN COOLING water supply system provides an adequate water supply for helium circulator water turbine operation and emergency cooling water to the steam generators for SAFE SHUTDOWN COOLING. The SAFE SHUTDOWN COOLING water supply system also supplies cooling water to other SAFE SHUTDOWN COOLING equipment via the firewater header. SAFE SHUTDOWN COOLING requirements are discussed in Section 1.4, 10.3, and 14.4 of the FSAR.

For protection against a single failure, there are two independent SAFE SHUTDOWN COOLING water flowpaths from the circulating water makeup storage ponds to the firewater header. There is a single pump pit for the three circulating water makeup pumps, and a separate pump pit for the two firewater pumps. The piping between the storage ponds and the pumps and the firewater header is redundant. The firewater header (10"L4502-D25 on PI 45) can be supplied from the firewater pumps via several different connections to ensure a reliable supply to SAFE SHUTDOWN COOLING equipment.

The SAFE SHUTDOWN COOLING equipment supplied from the firewater header includes the emergency condensate header (contained in Specification 3/4.5.4), the emergency feedwater header (contained in Specification 3/4.5.4), the instrument air compressors and after coolers (contained in Specification 3/4.7.3), the standby diesel generator coolers (contained in Specification 3/4.8.1), and the PCRV liner cooling system (contained in Specification 3/4.6.2).

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. The Specification limit of 20 million gallons of water is in addition to any ice formation. The firewater system has two redundant 100% capacity firewater pumps, each rated for 1500 gpm at 125 psig TDH. The main pump is electric-motor driven, and the standby pump is diesel-engine driven. With 370 gallons of fuel in storage, the diesel-engine driven fire pump can operate at rated conditions for 24 hours which is adequate time to have more fuel delivered to the site. Each pump has its own driver with independent power supplies and controls and is located in a separate room, divided by a 3-hour fire rated concrete wall.

The fire water pumps take suction from independent pits which are supplied from two storage ponds via circulating water makeup pumps.

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BASIS FOR SPECIFICATION LCO 3.5.5/SR 4.5.5 (Continued)

ACTIONS

With the CALCULATED BULK CORE TEMPERATURE greater than 760 degrees F, and with two out of three circulating water makeup storage pumps inoperable or with any one circulating water makeup pond inoperable or a firewater pump inoperable, a restoration time of 72 hours is considered sufficient, as this constitutes a loss of redundancy in SAFE SHUTDOWN COOLING equipment. 72 hours is consistent with Actions required in Light Water Reactors for loss of redundancy in Emergency Core Cooling System equipment. Any combination of components may be inoperable for 48 hours, provided a flow path is still available. However, with all circulating water makeup pumps, headers, or firewater pumps inoperable, a restoration time of 1 hour is specified, as all means of SAFE SHUTDOWN COOLING water supply are lost.

Also, with the CALCULATED BULK CORE TEMPERATURE below 760 degrees F, a single flowpath may be inoperable for 14 days or backup equipment may be placed in service. This allows periodic maintenance of components such as the storage ponds, when the decay heat removal requirements are reduced. With no flow paths, 12 hours are allowed to be in at least SHUTDOWN, consistent with the time allowed for an orderly shutdown from STARTUP. Then, as long as the inoperable equipment is restored to OPERABLE status prior to reaching 760 degrees F, no further Actions are required.

The surveillances identified in this specification will ensure that all equipment, water supplies, and flow paths will remain OPERABLE as specified in order to meet those SAFE SHUTDOWN COOLING requirements specified above.

The use of 760 degrees F CALCULATED BULK CORE TEMPERATURE as a division between the ACTIONS is explained in the BASIS for Specification 3.0.5.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of primary coolant helium flow.

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PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

PCRV SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.6.1.1 All PCRV safety valves shall be OPERABLE with:

a. Rupture disc and safety valve settings as specified in Table 2.2.1-1,

b. Both inlet block valves locked open,

c. Less than 5 psig between each rupture disc and relief valve,

d. Less than 20 psig in the safety valve containment tank, and

e. Less than 200 psig in the safety valve pilot stage bellows.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia

ACTION: With any PCRV safety valve 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.

SURVEILLANCE REQUIREMENTS

4.6.1.1 The PCRV safety valve installation shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the pressure between each rupture disc and relief valve, in the safety valve containment tank, and in the safety valve pilot stage bellows, is within the above limits.

b. 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 the applicable test interval specified in the ASME Code. Rupture disc assemblies shall be tested to verify settings. Safety valves shall be tested in accordance with applicable ASME Code requirements to verify setpoints.

c. The PCRV safety valve containment tank closure bolting shall be visually examined for absence of surface defects when the tank is opened for testing of the overpressure protection devices. Tank closure flange leak tightness shall be determined following tank closure.

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BASIS FOR SPECIFICATION LCO 3.6.1.1/SR 4.6.1.1

The PCRV safety valve installation (consisting of a containment tank containing two parallel trains, each of which has a manual block valve and a rupture disc upstream of the safety valve, the trains 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 ensure protection assuming a single failure. The manual block valves are locked open and their position is ensured by administrative controls. (FSAR Section 6.8.2.1). The position indication circuits associated with each block valve are functionally tested and calibrated when testing either train of the system.

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.

Pressure buildup between the rupture disc and safety valve indicates leakage past the rupture disc and shifts the effective set pressure of the rupture disc. The high-set rupture disc has a maximum allowable set pressure of 832 + 8 = 840 psig. To ensure that the high-set rupture disc fractures at less than or equal to the PCRV Reference Pressure (845 psig), the pressure buildup cannot exceed 5 psig. This value is also used for the low-set rupture disc to maintain the design safety margin. Verification that the interspace pressure is less than 5 psig may be made by absence of alarms or other means.

Although designed for 845 psig, the PCRV safety valve containment tank is normally at atmospheric pressure. The tank is provided with a pressure alarm (set at a nominal 15 psig) to detect any primary coolant leakage into the tank. Pressure in this tank does not effect safety valve operation.

Pressure buildup in a safety valve's pilot stage bellows is indicative of bellows leakage that would interfere with safety valve operation, and this pressure is alarmed. The actual setpoint for this alarm is arbitrary and an acceptance range has been established as 10-200 psig, to allow for switch inaccuracies and to accommodate the existing plant equipment.

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BASIS FOR SPECIFICATION LCO 3.6.1/SR 4.6.1 (Continued)

Any pressure buildup in either the PCRV safety valve containment tank or the rupture disc/safety valve interspaces can be remote-manually released to the reactor plant ventilation system, upstream of the exhaust filters. The appropriate action when any of the conditions are not met is to depressurize the PCRV which obviates the need for overpressure protection. The ACTION statement requires this after reasonable times for corrective action and SHUTDOWN.

PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

STEAM GENERATOR/CIRCULATOR PENETRATION OVERPRESSURE PROTECTION

LIMITING CONDITION FOR OPERATION

3.6.1.2 One PCRV penetration overpressure protection train protecting the penetrations for each steam generator loop and one train protecting each circulator penetration shall be OPERABLE with:

a. Rupture disc and safety valve settings as specified in Table 2.2.1-1,

b. The associated block valve open, and

c. Less than 5 psig between the rupture disc and the associated safety valve.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia

ACTION: With both PCRV penetration overpressure protection trains protecting the penetration(s) of any steam generator loop or circulator inoperable,

a. Within 12 hours either:

1. Restore one train to OPERABLE status, or

2. Depressurize the process piping in the affected penetrations to less than primary coolant pressure.

b. If neither of the above ACTIONS are performed within 12 hours, be in at least SHUTDOWN within the next 24 hours and depressurized to less than 100 psia within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The steam generator and circulator penetration overpressure protection trains shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying less than 5 psig between the rupture disc and safety valve.

b. At least once per 31 days by verifying the open position of the block valve to the OPERABLE overpressure protection train.

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 for each penetration interspace is tested in accordance with the test frequency specified in the ASME Code, at an approximate interval of 2 1/2 years. Safety valves shall be tested in accordance with the applicable ASME Code requirements to verify setpoints. In conjunction ith this test, one rupture disc from a steam generator penetration overpressure protection train and one rupture disc from a helium circulator penetration overpressure protection train shall be visually examined to verify that the membrane is free of defects and the knife blade remains sharp.

2. Performing a CHANNEL FUNCTIONAL TEST of the control, interlock, and indication circuits associated with each of the penetration overpressure protection assembly block valves.

BASIS FOR SPECIFICATION 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 and one valve closed at any time.

For the train to be considered OPERABLE, the rupture disc and safety valve must be set as specified in LSSS 2.2.1, the block valve must be open and the interspace between each rupture disc and the corresponding safety valve interspace must not be pressurized above 5 psig. Verification that the pressure is less than 5 psig may be made by absence of alarms or other means. Since the relief valves protect the penetrations from overpressurization resulting from rupture of high pressure process piping (e.g. bearing water, feedwater, emergency feedwater, steam) within the penetration, the appropriate action is to either depressurize the high pressure process piping or shutdown and depressurize the PCRV. The ACTION statement requires this within reasonable times considering the probability and risks of such an occurence.

The surveillances verify the settings of the safety valves, and ensure calibration and function of the pressure switches, alarms and position control and indication circuits. The intervals specified are suitable for the nature and reliability of system components.

Testing of a PCRV penetration overpressure protection assembly can be performed during plant operation since the assemblies are accessible and LCO 3.6.1.2 requires only one assembly to be OPERABLE at any time. The safety valve in each assembly is tested to demonstrate that it opens at the correct set pressure.

The rupture discs are not provided with a testable design feature, and therefore, cannot be tested. However one rupture disc of each type assembly is visually examined to verify that the membrane is free of defects and that the knife blade remains sharp.

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PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

INTERSPACE MINIMUM PRESSURIZATION

LIMITING CONDITION FOR OPERATION

3.6.1.3 The interspaces between the primary and secondary PCRV penetration closures shall be pressurized with purified helium gas as follows:

a. At a pressure greater than primary coolant pressure, or

b. With a leak pathway from one or more steam generator interspace(s) to the cold reheat steam system, 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 interspace(s) below the minimum required pressure, restore the pressurization to within its limit 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

4.6.1.3 The PCRV penetration interspaces shall be verified to be within the above limits at least once every 24 hours by verifying that the PCRV penetration interspace supply header pressures are above primary coolant pressure, and that the steam generator penetrations in affected loops are above cold reheat steam pressure.

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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) ensures 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 these interspaces into the cold reheat steam system piping within the interspaces. In this case, the steam generator penetration interspaces in the affected loop may be maintained at a pressure less than primary coolant pressure, but slightly greater than cold reheat steam 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 past the air ejector monitor and then through the monitored plant exhaust stack. Consequently, more stringent leak testing and interspace 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 protect against the leakage of unacceptable quantities of contaminated helium gas. The condenser air ejector monitor, the reactor building ventilation system monitors, and radioactive gaseous effluents from the plant are addressed and controlled by Specification 8.0, "Radiological and Environmental Technical Specifications".

While interspace pressure below the minimum required pressure, in itself, poses no immediate hazard, leakage through both the primary and secondary closures, could lead to primary coolant leakage from the PCRV to the reactor building atmosphere. However, other specifications ensure that such leakage would be limited to an acceptable rate and would be detected, filtered, monitored and exhausted from the ventilation exhaust stack. The appropriate action is to restore interspace pressurization to the necessary level in a reasonable time (24 hours) or be in SHUTDOWN within the next 24 hours and depressurize the PCRV.

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BASIS FOR SPECIFICATION LCO 3.6.1.3/SR 4.6.1.3 (Continued)

Maximum PCRV and interspace pressures are limited to 845 psig per SL 2.1.2.

The surveillances provide for monitoring the penetration interspace supply pressure. The SURVEILLANCE INTERVAL is suitable for the type and reliability of the components involved. Interspace pressure monitoring instrumentation is tested and calibrated in accordance with administrative procedures.

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3/4.6.1 PCRV PRESSURIZATION

PCRV CLOSURE LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.4 The helium leakage through PCRV penetration closures shall be maintained within the following limits:

a. Total helium leakage through all the primary closure seals in any of penetration groups I through VIII shall not exceed an equivalent leak rate of 400 pounds per day at a differential pressure of 10 psi, and

b. Total helium 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. Total helium leakage from steam generator penetration group III or IV to the reheat steam system shall not exceed 700 pounds per day.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia

ACTION: With PCRV penetration interspace helium leakage exceeding the above limits, reduce the helium leakage to within 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 PCRV penetration interspace leakage shall be determined to be within the above limits at least once per 92 days or within 24 hours after an unanticipated increase in pressurization gas flow, by determining the leak rate of the PCRV penetration primary and secondary closures, and the leak rate of steam generator penetration interspace to the reheat steam system.

BASIS FOR SPECIFICATION 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 purified 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 purified helium.

The normal gas supply to all the penetration closure interspaces is from the helium purification system and is continuously monitored for flow so that an increase in closure leakage can be sensed and alarmed. The penetration closure interspaces are supplied with pressurizing gas in groups through the arrangement of the purified helium piping. The grouping of the penetrations, defined in FSAR Section 5.12.2, is as follows:

Group I: All penetrations in the top head of the PCRV (37 - control rod drive, 2 - high temperature filter-absorber, and 1 - top access).

Group II: All instrument penetrations (20) plus the bottom access penetration.

Group III: The 6 steam generator penetrations, Loop I.

Group IV: The 6 steam generator penetrations, Loop II.

Group V-VIII: Each helium circulator penetration.

The leak rate limitations for the primary closures are based on a differential pressure of 688 psi, which would be the differential pressure across a primary closure in the event a secondary closure should fail. The calculated permissible leakage rate across the primary closure would be in excess of 1145 pounds per hour at a differential pressure of 688 psi. Converting the 1145 pounds per hour leak 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). This leakage flow can readily be detected on the pressurizing gas flow indicator. It is assumed that under these conditions, the entire inventory of primary coolant would leak through the primary closure. (The associated activity release would be similar to that release resulting from the maximum credible accident (MCA) discussed in Section 14.8 of the FSAR). Assuming design primary coolant activity, and assuming a dilution factor of 2.7 x 10 -3 sec/m3, the resultant dose is at least an order of magnitude less than the limits of 10 CFR 100 at the exclusion area boundary.

BASIS FOR SPECIFICATION LCO 3.6.1/SR 4.6.1 (Continued)

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 ensure 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. Cold reheat steam pressure varies with plant load, but will generally be at least 50 psi below primary coolant 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 allowable leakage of 700 pounds per day of helium from any steam generator penetration group to the reheat steam system is based on maintaining a satisfactory main condenser vacuum. Since this leakage is normally purified helium, there is normally no radiological impact. Monitoring by the reheat loop activity monitor(s) per Specification 3.6.1.5 ensures the absence of radioactive effluent. The air ejector monitor serves as a backup to the reheat loop activity monitors. The air ejector monitor, the reactor building ventilation system monitors, and radioactive gaseous effluents from the plant are addressed and controlled by Specification 8.0, "Radiological and Environmental Technical Specifications".

In the determination of closure leakage at the reference differential pressure, laminar leakage flow is conservatively assumed. Therefore, in correcting the determined closure leakage to reference differential pressure, the ratio of the reference differential pressure and test differential pressure is used.

If helium leakage exceeds the limits, the appropriate action is to take corrective action within a reasonable time or SHUTDOWN and depressurize the PCRV.

BASIS FOR SPECIFICATION LCO 3.6.1/SR 4.6.1 (Continued)

The surveillances provide for measuring closure leak rates at surveillance intervals suitable for the type of closures.

Additional information on the BASIS, can be found in the references listed below.

References

1. Letter P-82007 from O.R. Lee (PSC) to Robert Clark (NRC), subject: Technical Specification Change LCO 4.2.7 and LCO 4.2.8 and Basis, dated January 8, 1982.

2. Letter G-82079 from George Kuzmycz (NRC) to O.R. Lee (PSC), subject: Fort St. Vrain Nuclear Generating Station, Amendment Number 26 to Facility Operating License Number DPR-34, dated March 18, 1982.

3. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 26 to Facility Operating License Number DPR-34, Public Service Company of Colorado, Fort St. Vrain Nuclear Generating Station, Docket Number 50-267, dated March 18, 1982.

4. FSAR Sections 5.8.2.3, 5.8.2.5.4, and 5.12.2.

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PCRV AND CONFINEMENT SYSTEMS

3/4.6.1 PCRV PRESSURIZATION

STEAM GENERATOR INTERSPACE RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.6.1.5 a. The gross activity monitors (channel 2263 and/or 2264) for the applicable steam generator penetration interspace group(s) (III and/or IV) shall be OPERABLE and aligned to monitor the applicable steam generator interspace(s).

b. The gross activity indication for the applicable steam generator interspace group(s) shall not exceed 200 cpm.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia, and when any steam generator penetration interspace group(s) (III and/or IV) is being maintained below primary coolant pressure. (Per Specification 3.6.1.3).

ACTION:

a. With the gross activity monitor for the affected steam generator interspace group III and/or IV inoperable:

1. At least once per 8 hours, take a grab sample of the applicable steam generator penetration interspace group(s) and analyze it for the noble gas gross activity within 24 hours, and

2. Restore the required channel(s) 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.

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b. With indication on the gross activity monitor for affected steam generator interspace group(s) exceeding 200 cpm, determine that the leak tightness of the affected steam generator penetration interspace primary closures meets the requirements of Specification 3.6.1.4 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

4.6.1.5 a. At least once per 12 hours, activity monitor channel 2263 or 2264 shall be monitored when its associated steam generator penetration interspace group is being maintained below primary coolant pressure, to determine that the indication does not exceed the above limit.

b. Activity monitor channel 2263 or 2264 shall be demonstrated OPERABLE when its associated steam generator penetration interspace group is being maintained below primary coolant pressure:

1. At least once per 31 days by:

   a) Performing a CHANNEL FUNCTIONAL TEST, and

   b) Verifying by sample flow or other means, 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.

2. At least once per 18 months by:

   a) Performing a CHANNEL CALIBRATION and,

   b) Cycling each power operated valve in the sample flow path through at least one complete cycle of full travel.

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BASIS FOR SPECIFICATION LCO 3.6.1.5/SR 4.6.1.5

Monitoring of the steam generator penetration interspace group III and/or IV 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, as addressed by Specification 3.6.1.3.

Penetration groups are defined in FSAR Section 5.12.2, and in the BASIS for Specification 3/4.6.1.4.

In the event that the monitors become inoperable, grab samples are an acceptable substitute for the monitors for a limited time. Alternatively, steam generator penetration interspace pressure could be increased to above primary coolant pressure, which would void the applicability of the specification.

If the interspace gross activity reaches the specified limits, which are about 200% of the normal background level, that would be indicative of primary closure leakage, which is addressed by Specification 3.6.1.4, and the primary closure leak rate shall be determined within the time specified in SR 4.6.1.4.b.

The surveillances ensure that the instrumentation is functional, accurate, and properly aligned. They also ensure that the steam generator penetration interspaces are monitored for activity on a frequent basis. The frequency of grab sample analysis is the same as other Specifications monitoring coolant activity (e.g., 4.4.1 a).

The condenser air ejector exhaust radiation monitor provides a backup means of detecting a steam generator penetration primary closure leak accompanied by a leak path into the reheat system. The reactor building ventilation system radiation monitors, in turn, serve as backups to the air ejector monitor. The air ejector monitor, the reactor building ventilation system monitors, and radioactive gaseous effluents from the plant are addressed and controlled by Specification 8.0, "Radiological and Environmental Technical Specifications".

Additional information on the above, can be found in FSAR Sections 7.3.5 and 11.1.1.

PCRV AND CONFINEMENT SYSTEMS

3/4.6.2 REACTOR PLANT COOLING WATER/PCRV LINER COOLING SYSTEM - OPERATING

LIMITING CONDITION FOR OPERATION

3.6.2.1 The Reactor Plant Cooling Water (RPCW)/PCRV Liner Cooling System (LCS) shall be OPERABLE with:

a. Two loops operating each with at least one heat exchanger and one pump operating,

b. At least three out of any four adjacent tubes on the core support floor side wall, core support floor bottom casing, PCRV cavity liner sidewalls and PCRV cavity liner bottom head operating,

c. At least five out of any six adjacent tubes on the PCRV cavity liner top head and core support floor top casing operating,

d. Tubes adjacent to a non-operating tube operating, and

e. The temperature rise for each operating LCS tube within the acceptance criteria of Specification 4.6.2.1.b.

APPLICABILITY: POWER, LOW POWER, STARTUP*, SHUTDOWN*, and REFUELING*

ACTION

a. With only one RPCW/PCRV LCS loop operating, ensure both heat exchangers are operating in the operating loop, restore the second loop to operating within 48 hours or be in SHUTDOWN within the following 12 hours and suspend all operations involving control rod movements resulting in positive reactivity changes.

b. With no heat exchangers in the operating loop operating or with no liner cooling system loop flow be in SHUTDOWN within 15 minutes and suspend all operations involving control rod movements resulting in positive reactivity changes.

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* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

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SPECIFICATION LCO 3.6.2.1 (Continued)

c. With less than the above required number of PCRV Liner Cooling System tubes operating, other than during single loop operation as permitted in ACTION a. above, restore the required tubes to operating status within 24 hours or be in SHUTDOWN within the following 24 hours and suspend all operations involving positive reactivity changes.

d. With any Liner Cooling System tube temperature rise greater than the acceptance criteria of SR 4.6.2.1.b below, restore the temperature rise to acceptable status within 7 days or submit a Special Report per Specification 6.9.2 within the next 14 days, identifying the cause of the excessive temperature rise, the corrective actions planned to restore the temperature rise to acceptable status, and the justification for continued operation with the existing temperature rise condition.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The RPCW/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:

1. Verifying that the average temperature rise for each LCS tube does not exceed the following limits. The tube temperature rise is the difference between the LCS tube outlet temperature and the respective inlet header temperature for an operating loop, and the average temperature rise is the average of the tube temperature rises for any given tube and the adjacent operating tubes on either side of it (three tubes total, unless otherwise indicated):

a) Top Head Penetrations

The average temperature rise for any tube in this area shall not exceed 30 degrees F.

b) Core Outlet Thermocouple Penetrations

The average temperature rise for any tube in this area shall not exceed 25 degrees F.

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SPECIFICATION LCO 3.6.2.1 (Continued)

c) Core Barrel Seal/Core Support Floor Area

The average temperature rise for tubes F7T43, F6T44, F11T45, F12T46, F5T47, and the adjacent tubes on either side of this tube group (seven tubes total) shall not exceed 47 degrees F.

d) Peripheral Seal Area

The average temperature rise for any tube in this area shall not exceed 28 degrees F.

e) The average temperature rise for any tube outside of the above identified hot spot areas shall not exceed 25 degrees F.

If the tube temperature rise for any LCS tube is not available due to an instrument failure, the tube temperature rise of the tube with an instrument failure may be inferred from the average of the tube temperature rises of two tubes on both sides (4 tubes total).

2. Functionally testing each reactor plant cooling water pump.

c. At least once per 12 months or at the next scheduled plant shutdown if not performed during the previous 12 months, by verifying the performance and mechanical condition of each RPCW pump.

d. At least once per 18 months by:

1. Performing a LCS redistribute mode functional test to verify the capability of rerouting most of the cooling water to the upper side walls and the top head.

2. Performing a functional test to verify the capability to increase the LCS surge tank pressure to 30 psig by adding helium to the tank.

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PCRV AND CONFINEMENT SYSTEMS

3/4.6.2 REACTOR PLANT COOLING WATER/PCRV LINER COOLING SYSTEM - SHUTDOWN

LIMITING CONDITIONS FOR OPERATIONS

3.6.2.2 The Reactor Plant Cooling Water (RPCW)/PCRV Liner Cooling System (LCS) shall be OPERABLE with one RPCW/PCRV LCS loop operating with at least one heat exchanger and one pump in each loop operating.

APPLICABILITY: STARTUP*#, SHUTDOWN*#, and REFUELING*#

ACTION: With no RPCW/PCRV LCS loop operating,

a. And with forced circulation cooling maintained, be in at least SHUTDOWN within 12 hours and restore at least one loop to operating status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, or suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

b. And with no forced circulation cooling, be in at least SHUTDOWN within 10 minutes, and restore at least one loop to operating status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, or suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The RPCW/PCRV LCS shall be demonstrated OPERABLE by performance of the requirements identified in SR 4.6.2.1.

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* Whenever CALCULATED BULK CORE TEMPERATURE is less than or equal to 760 degrees F.

\# The core support floor sidewall zone of the PCRV Liner Cooling System may be valved out when PCRV pressure is less than or equal to 150 psia and CORE AVERAGE INLET TEMPERATURE is less than or equal to 250 degrees F

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BASIS FOR SPECIFICATION LCO 3.6.2/SR 4.6.2

During operation at POWER, two PCRV liner cooling system loops are required to maintain PCRV liner cooling system temperatures and PCRV stresses within the FSAR design criteria (FSAR Section 5.9.2., Thermal Barrier and Liner Cooling System Design and Design Evaluation). Analysis supporting the PCRV Liner Cooling System design (FSAR Section 5.9.2.4) demonstrates that operation at full power with one cooling loop for 48 hours satisfies the design criterion of a maximum temperature increase of 20 degrees F in the bulk temperature of the PCRV concrete within one and one-half feet of the liner. Operation on one loop during a loss of forced circulation accident using a PCRV liner cooldown with an increased liner cooling water system cover pressure of 30 psig may result in temperature rises across individual cooling tubes of 240 degrees F (outlet temperature of approximately 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).

The liner cooling tubes are spaced in such a manner as to limit local concrete temperatures adjacent to the liner to 150 degrees F under normal full power design operating conditions. However, potential failures of cooling tubes were analyzed and their limits follow.

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 adjacent tubes on the PCRV cavity side walls, PCRV cavity bottom casing, core support floor side wall or core support floor liner bottom head, or a single tube in any set of six adjacent tubes on the PCRV cavity liner top head and core support floor top casing. A failed tube which doubles back on itself is considered a single tube failure. In these cases, the local temperature in the concrete would be less than 250 degrees F (during normal two loop operation), an acceptable concrete temperature (FSAR 5.9.2.3.).

Operation of the PCRV liner cooling system during startup testing disclosed hot spots. These locations were identified and analyzed in FSAR Section 5.9.2.8. The engineering evaluation indicated that operation with the hot spots would not compromise PCRV integrity and continued operation is acceptable. Four of the seven hot spots have liner cooling tubes which are expected to have temperature rises greater than 20 degrees F, as identified in FSAR Section 5.9.2.8, Table 5.9-1 and Table 5.9-4.

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BASIS FOR SPECIFICATION LCO 3.6.2/SR 4.6.2 (Continued)

Tube temperature rises are limited on an overall basis by the 20 degree system temperature rise limitation of LCO 3.6.3.d. The Liner Cooling System has been analyzed for failed tubes and for localized hot spots, with the conclusion that the integrity of the PCRV is not compromised by these conditions (FSAR 5.9.2.8). In the hot spot analysis cited as Reference 15 in FSAR Section 5.9, Los Alamos National Laboratories (LANL) indicated that with tube temperature rises up to 47 degrees F, the concrete temperatures could be as high as 354 degrees F and the resulting stresses were below allowable limits of the ASME Code. LANL considered that other hot spots with lower temperatures are acceptable without further explicit analysis. LCS tube temperature rises of 25 degrees F result in stresses less severe than those evaluated by LANL, and are acceptable without further analysis.

Based on the LANL conclusions, individual tube temperature rises are permitted to exceed the 20 degrees F limit if the average of three adjacent tube temperature rises does not exceed 25 degrees F. This provision assures relatively low heat input to the concrete while allowing for small variations in individual tube flow rates during plant operation. Also, the average temperature rise limits for the identified hot spots assures that the heat input to the concrete in these areas is substantially the same as that analyzed, while allowing for local minor variations in LCS flow rates and temperature distribution during plant operation.

The use of an average tube temperature limit is acceptable because non-uniform temperature distributions in the PCRV concrete result in some minor variations in individual liner cooling tube temperature rises. These non-uniform temperatures result in thermal stresses in the structure, but these stresses tend to relax due to creep and other inelastic effects, particularly in areas of local stress concentration. Therefore, only the bulk temperature of the PCRV concrete is significant in establishing the thermal loading of the PCRV and averaging the temperature rise in three adjacent tubes is justified.

The ACTION times specified for recovery of two operating loops come from analyses described in FSAR Section 5.9.2.4, i.e., 48 hours operation on one loop before temperature of the bulk concrete would rise 20 degrees F. With the number of cooling tubes less than required, a 24 hour action time is sufficient to identify and restore the tube to operating status (if possible) or SHUTDOWN to make permanent repairs.

With any LCS tube temperature rise greater than the acceptance criteria, 7 days are allowed to restore it as the temperature effects on PCRV concrete occur slowly. Also, the preparation of a report if the temperature rise is not restored within limits assures an evaluation of continued operability in that condition.

BASIS FOR SPECIFICATION LCO 3.6.2/SR 4.6.2 (Continued)

The surveillance(s) and their respective intervals are specified to verify operability of the liner cooling system. Components and features of the reactor plant cooling water system that are not safety related do not affect LCS operability. The ISI/IST program at Fort St. Vrain verifies OPERABILITY of those barriers that separate safety and non-safety related portions of the system. A 24 hour surveillance on system flow rates provides additional verification of flow as process alarms monitor flow continuously in each liner cooling loop. Indiv'dual tube failures would be expected to occur slowly, thus a 31 day SURVEILLANCE INTERVAL will detect tube failures in time to take corrective action.

With CALCULATED BULK CORE TEMPERATURE less than or equal to 760 degrees F, one operating liner cooling system loop is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

When the PCRV pressure is less than 150 psia and CORE AVERAGE INLET TEMPERATURE is less than 250 degrees F, the core support floor sidewall zones of the liner cooling system may be valved out as concrete temperatures will be less than the 250 degree FSAR limitation. Thus, leaking liner cooling tubes which are awaiting repairs will not contribute to potential moisture ingress into the primary system.

In Surveillance Requirement 4.6.2.1.b., tube outlet temperatures are determined by thermocouple readings. In the event of an instrument failure (i.e., a thermocouple is thought to be failed), the tube with the failed thermocouple may be considered OPERABLE if thermocouple readings for two adjacent tubes on either side of that tube are within their respective temperature limits. If the tube itself failed rather than the thermocouple, then the temperature of adjacent tubes would be expected to rise. Thus, a failed thermocouple can be identified vs an actual tube failure. Power operation may continue until such time as the thermocouple can be repaired or replaced as long as the total of four adjacent tubes (two on either side of the tube with the failed instrument) are within their respective temperature limits.

The use of 760 degrees F CALCULATED BULK CORE TEMPERATURE as a division between the APPLICABILITY of Specification 3.6.2.1 and 3.6.2.2 is explained in the Basis for Specification 3.0.5.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, or primary coolant helium flow.

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3/4.6.3 REACTOR PLANT COOLING WATER/PCRV LINER COOLING SYSTEM TEMPERATURES

LIMITING CONDITIONS FOR OPERATION

3.6.3 The RPCW/PCRV Liner Cooling System (LCS) 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 outlet temperature shall not exceed 120 degrees F.

c. The maximum change of the weekly average PCRV concrete temperature shall not exceed 14 degrees F per week.

d. The maximum temperature difference across the RPCW/PCRV Liner Cooling Water Heat Exchanger (LCS portion) shall not exceed 20 degrees F.

e. The minimum average LCS water temperature shall be greater than or equal to 100 degrees F.

APPLICABILITY: At all times

ACTION:

With any of the above limits not satisfied, restore the limit(s) within 24 hours, or be in SHUTDOWN or REFUELING within the next 24 hours, and suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

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SURVEILLANCE REQUIREMENTS

4.6.3 The RPCW/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. The weekly average water temperature is determined by computing the arithmetic mean of 7 temperatures, representing each of the last 7 days of common PCRV cooling water outlet temperatures in each loop. Each day results in a new computation of a weekly average water temperature. The new weekly average is then compared to the weekly average water temperature computed 7 days earlier to verify the limit of Specification 3.6.3.c.

d. Verifying that the maximum delta T across the RPCW/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 average 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 (LCS side) inlet and outlet temperatures.

BASIS FOR SPECIFICATION LCO 3.6.3/SR 4.6.3

The temperature limits of this Specification are consistent with the general design limits of the liner and PCRV concrete, as provided in FSAR Sections 5.9, 5.7, 5.12 and 9.7. The PCRV liner and its associated cooling system assist in maintaining 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 ensure that thermal stresses on the PCRV concrete and liner are within FSAR analyses described above and that PCRV integrity is maintained.

Since the PCRV concrete has a large thermal mass and inertia, temperatures respond very slowly to any changes in the specified parameters. A 24 hour restoration and ACTION time is consistent with the expected slow temperature response of the PCRV. As a precaution, the plant would be SHUTDOWN and/or remain in REFUELING mode until temperatures were stabilized.

PCRV AND CONFINEMENT SYSTEMS

3/4.6.4 PCRV INTEGRITY

STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.4.1 The PCRV structural integrity shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.4.1.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia

ACTION: With the PCRV structural integrity not in compliance with the above requirement, restore PCRV integrity within 24 hours or be SHUTDOWN within the subsequent 24 hours and depressurized to below 100 psia within the next 30 hours.

SURVEILLANCE REQUIREMENT

4.6.4.1.a The structural integrity of the PCRV with regard to the tendons shall be demonstrated:

1. At least once per 12 months by:

   a) Visually examining the Control, New, and Worst-Case tendon anchor assemblies specified in Table 4.6.4-1 including tendon end caps, tendon wire button-heads, anchor/bushing assemblies, stressing washers, shims, bearing plates, and visually verifying the integrity of the anchor assembly, checking for raised button heads, general corrosion, concavity, water or other apparent failures, and examining the surrounding concrete for structural deterioration, and

   b) Performing lift-off tests to determine the load carried by the control tendons identified in Table 4.6.4-2, and verifying that the loads exceed the minimum allowable values given in Table 4.6.4-3.

2. At least once per 24 months, by performing lift-off tests to determine the load carried by the New tendons* identified in Table 4.6.4-2, and verifying that the loads exceed the minimum allowable values given in Table 4.6.4-3.

3. Acceptance Criteria:

   Engineering evaluations of the above test results shall be made as the tendon inspection program progresses with the intent of ensuring that the prestressing system is performing its design function. Specific engineering evaluations shall be mandatory for any circumferential barrel tendon with greater than or equal to 23 failed wires** and for any tendon in any of the remaining tendon groups with greater than or equal to 34 failed wires, or any tendons not meeting the minimum lift-off load requirements of Table 4.6.4-3.

4. Reporting:

   Special Reports, summarizing the status and results of this inspection/testing program, shall be submitted every twelve months to the Commission.

* Lift-off tests for tendons that have not previously been lifted off would include removal of the shim plates to permit visual examination and as necessary reapplication of grease to accessible areas of the tendon. Repetitive lift-off tests on the same tendon not in a control group may not include removal of shim plates for visual examination. Lift-off tests for tendons in designated control groups will include removal of shim plates and visual examination.

** A failed wire is one that is broken and carries no load, such as one with a raised button head or as may have been previously identified as failed by visual inspection. For tendons that are not accessible on both ends, the total number of assumed failed wires shall be 1.2 times the number of failed wires identified on the accessible end.

SURVEILLANCE REQUIREMENTS

4.6.4.1.b Continuous monitoring of the structural integrity of the PCRV tendons shall be demonstrated by performance of the following PCRV Tendon Load Cell surveillances:

1. At least once per 12 months, by performing a CHANNEL FUNCTIONAL TEST of the load cell alarm circuit between the data acquisition system room and the control room.

2. At least once per 5 years for representative load cells, and whenever a lift-off test is performed on a tendon with a load cell, and whenever a tendon with a load cell is detensioned, by checking for shift in load cell reference points.

3. Any degradation of the tendon load cell readings detected during the tests required by 4.6.4.1.b.2 shall be subjected to an engineering evaluation.

   Any abnormal degradation that the evaluation identifies shall be reported to the Commission on a preliminary basis within 30 days after the finding. A final Special Report shall be submitted to the Commission pursuant to Specification 6.9.2. This report shall include a description of the degradation, the inspection procedures and results, and the corrective actions taken, if any.

4.6.4.1.c The structural integrity of the PCRV concrete shall be verified as follows:

1. At least once per 5 years, beginning with the fourth refueling outage, demonstrate PCRV concrete integrity by recording PCRV deformations and deflections at vessel midheight and at the center of the top head, during a vessel pressurization to operating pressure.

2. At least once per 5 years, beginning 3 years following initial power operation, demonstrate PCRV concrete integrity by measuring its permeability to verify that the flow rate through each sample tube is not more than four times the baseline value or 40 lb/hr, whichever is greater.

3. At least once per 10 years, beginning 3 years following initial power operation, demonstrate PCRV concrete integrity by mapping crack patterns on the visible surface of the PCRV and comparing them with previously recorded cracks. Concrete cracks which exceed 0.015 inches in width have been recorded. Recorded cracks shall be assessed for changes in length and width and any new cracks exceeding 0.015 inches in width shall be recorded.

4. At least once per 10 years, the PCRV support structure shall be visually examined for evidence of structural deterioration.

5 Any degradation of the PCRV concrete detected during the above required tests shall be subjected to an engineering evaluation.

   Any abnormal degradation that the evaluation identifies shall be reported to the Commission on a preliminary basis within 30 days after the finding. A final Special Report shall be submitted to the Commission 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, if any.

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	5	3	2
Longitudinal	18	12	6

WORST-CASE TENDONS: VM-30, BILU3, BILU4, CO 2.5, CM 4.6

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" shall consist of a tendon population selected at random for visual inspection over the next specified surveillance period. Selection shall be such that the total population of accessible tendons in that group shall be inspected before beginning any repeat inspection.

3. The "Total Number of Control Tendons" shall consist of the same population of tendons in each tendon group that shall be selected and shall remain constant for all inspection surveillance cycles. The criteria for selection of these tendons shall be to select those tendons which are accessible and have corrosion.

* 39 New tendons shall be inspected until all remaining accessible circumferential tendons without a surveillance on at least one end since March 1, 1984, have been surveilled on at least one end. After this, the total number of New tendons is 13.

TABLE 4.6.4-2

TENDON LIFT-OFF 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	3	2	1
Longitudinal	11	8	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 shall have only the top end-anchor assembly lifted off.

2. The "Total Number of New Tendons" shall consist of a tendon population selected at random for lift-off testing over the next specified surveillance period. Selection shall be such that the total population of accessible tendons in that group shall be tested before beginning any repeat 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 shall be selected and shall remain constant for all test surveillance cycles. The criteria for selection of these tendons shall be to select those tendons which are accessible and have corrosion. These tendons shall 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 shall be a tendon monitored by a load cell.

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TABLE 4.6.4-3

TENDON LOAD REQUIREMENTS

TENDON GROUP	MINIMUM TENDON LOADS (1) (kips)
Circumferential:	
at Heads	765
at Barrel	935
Top Cross Head	725
Bottom Cross Head	725
Longitudinal	1090

NOTES

1. Minimum tendon loads at jacking end.

PCRV AND CONFINEMENT SYSTEMS

3/4.6.4 PCRV INTEGRITY

LINER

LIMITING CONDITION FOR OPERATION

3.6.4.2 The PCRV Liner integrity (primary coolant boundary) shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.4.2.

APPLICABILITY: Whenever PCRV exceeds 100 psia

ACTION: With the PCRV Liner integrity not in compliance with the above requirement (without primary coolant boundary integrity), restore PCRV Liner integrity within 1 hour or depressurize to below 100 psia within the next 30 hours.

SURVEILLANCE REQUIREMENT

4.6.4.2 The integrity of the PCRV Liner shall be verified as follows:

a. During the fifth REFUELING CYCLE and every tenth REFUELING CYCLE thereafter by:

   1. Removing and testing 3 sets of 12 specimens with dosimeters from the 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.

   2. The testing program shall meet the requirements of ASTM-E-185-70 with the following exceptions:

      a) Tensile specimens are not required.

      b) Thermal control specimens are not required.

b. At least once per 10 years, beginning 5 years following initial power operation, by examining the PCRV liner for corrosion induced thinning using ultrasonic inspection techniques. Thinning less than or equal to 10% is acceptable.

c. At least once per 12 months by performing a CHANNEL CALIBRATION of the instrumentation which monitors and alarms pressure in the core support floor and core support floor columns.

d. Any degradation of the liner detected during the above required tests shall be subjected to an engineering evaluation.

Any abnormal degradation that the evaluation identifies shall be reported to the Commission on a preliminary basis within 30 days after the finding. A final Special Report shall be submitted to the Commission pursuant to Specification 6.9.2. This report shall include a description of the degradation, the inspection procedures and results, and the corrective actions taken, if any.

LIMITING CONDITION FOR OPERATION

3.6.4.3 a. The integrity of the PCRV penetrations and wells shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.4.3.

b. Valves used for automatic isolation of portions of the RPCW system and the purification cooling water system that may be required for confinement of primary coolant shall be OPERABLE.

APPLICABILITY: Whenever PCRV pressure exceeds 100 psia

ACTION: a. With the integrity of the PCRV penetrations and wells not in compliance with the above requirement, restore the required integrity within 24 hours or be SHUTDOWN within the subsequent 24 hours and depressurized to below 100 psia within the next 30 hours.

b. With any of the above required isolation valves inoperable, restore the inoperable valves to OPERABLE status within 72 hours or be in at least SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENT

4.6.4.3 PCRV penetration, well, and isolation valve integrity shall be demonstrated:

a. Prior to pressurization above 100 psia by checking all PCRV primary and secondary closures for proper integrity by ensuring that all bolts, cover plates, seals, etc., are installed.

b. At least once per 6 months by at least partially stroking the above required isolation valves that are capable of being tested.

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c. At least once per 12 months or at the next scheduled plant shutdown if not performed during the previous 12 months, by functionally testing the above required isolation valves.

d. At least once per 18 months by:

1. Functionally testing automatic isolation valves associated with pressurizing, purging, and venting PCRV penetration interspaces to verify that they isolate as designed.

2. Visually examining and verifying that the accessible portions of the refueling penetration hold-down plate bolting show no abnormal degradation.

3. Performing a CHANNEL FUNCTIONAL TEST of the instruments, controls, and interlocks used to automatically isolate each purification system,

4. Performing a CHANNEL CALIBRATION of the associated pressure switch and alarm for each helium purification cooler well, and

5. Determining closure leak tightness for each helium purification cooler well.

e. At least once per 5 years by:

1. Surface examination by Magnetic Particle Test (MT) or Dye Penetrant Test (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 for abnormal degradation 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 High Temperature Filter Absorber (HTFA) penetration purge lines to ensure they close as designed.

4. Testing the check valves integral with HTFA or refueling penetrations, only when such a penetration is open for refueling or maintenance, if the check valves have not been tested in the last 5 years, to ensure they function as designed.

5. Functionally testing the controls, position indication, and fail safe operation for remote manual isolation valves associated with pressurizing, purging, and venting PCRV penetration interspaces, to verify they operate as designed.

f. At least once per 10 years by:

1. Surface examination (MT or PT) of accessible portions of the following 2 welds in the bottom access penetration:

   a) The penetration shell-to-spherical head weld, and

   b) The spherical head-to-closure flange weld.

2. Visually examining for surface defects the accessible portions of the bottom access penetration primary closure split ring assembly and its secondary closure bolting, and

3. Visually examining 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 of the bolting attaching the support skirt to the PCRV outer wall.

g. Any degradation of the penetrations detected during the above required tests shall be subjected to an engineering evaluation.

Any abnormal degradation that the evaluation identifies shall be reported to the Commission on a preliminary basis within 30 days after the finding. A final Special Report shall be submitted to the Commission pursuant to Specification 6.9.2. This report shall include a description of the degradation, the inspection procedures and results, and the corrective actions taken, if any.

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BASIS FOR SPECIFICATION LCO 3.6.4/SR 4.6.4

The limitations of this specification ensure that the structural and primary coolant boundary integrity of the Prestressed Concrete Reactor Vessel (PCRV) will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure the concrete remains in net compression up to the REFERENCE PRESSURE of 845 psig. This pressure is in excess of the highest setpoint of the two PCRV relief trains.

If PCRV integrity cannot be restored within 24 hours, orderly shutdown within the subsequent 24 hours as required and reactor depressurization within the next 30 hours allow for the removal of decay heat and for excess primary coolant helium to be purified and removed to the helium storage system. The deterioration of PCRV integrity is an extremely slow process (e.g., caused by corrosion). Therefore, an orderly shutdown and depressurization is appropriate, considering the extremely large factors of safety in the PCRV design.

Various monitoring programs are specified to ensure the PCRV structural and primary coolant boundary integrity. These are described in Sections 5.13.4 and 5.13.8 of the FSAR. They consist primarily of monitoring and evaluating the PCRV with respect to tendon corrosion, tendon load cell surveillance, liner surveillance, and concrete structure surveillance. Details of each program are identified in the appropriate Surveillance Requirement. Additional information on the monitoring instrumentation sensors is described in Appendix E-17 of the FSAR.

PCRV Tendons (SR 4.6.4.1.a)

This Surveillance Requirement monitors the prestressing system and develops a tendon corrosion data base.

Visual and lift-off examinations of tendon assemblies ensure that the prestressing system has not degraded. Quantities and frequency of inspections and tests were agreed to in discussions with the Commission.

The substantial increase in the surveillance program together with a very high percentage of tendons that have already been inspected ensure that the PCRV prestressing system is capable of performing its design function.

The basis for the minimum allowable tendon loads of Table 4.6.4-3 is included in Reference 3.

The 23 failed wires are equal to 15% of the wires in the circumferential barrel tendons, and the 34 failed wires are equal to 20% of the wires in tendons in the other tendon groups. The basis for the engineering evaluations/failed wire criteria is included in Reference 1.

Tendon Load Cells (SR 4.6.4.1.b)

Since the relationship between effective prestress and PCRV internal pressure is directly related and easily calculated, monitoring tendon loads is a reliable means of ensuring that the vessel is capable of containing its design pressure.

Monitoring tendon load changes ensures that corrosion, concrete strength reduction, or excessive steel relaxation have not occurred to the extent that they would compromise PCRV integrity. Load changes, as reflected by load cells, are monitored in the control room by an alarm system which alerts the reactor 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). (FSAR Section 5.13.8).

Concrete Integrity (SR 4.6.4.1.c)

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. Larger 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. Initial crack mapping was performed prior to and following the Initial Proof Test Pressure (IPTP), and subsequent inspections were performed after the end of the first and third years following initial power operation. 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.

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 be made for evidence of structural deterioration. Significant cracking patterns or sizes will be subjected to an engineering evaluation with respect to their impact on the integrity of the PCRV.

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. Concrete permeability is determined by measuring flow from a test gas bottle, through sample tubes that have been installed at several locations into the concrete. An acceptable flow rate of four times the baseline value or 40 lb/hr, whichever is greater, is based on acceptance criteria developed previously as part of the ISI procedures.

The interval for the above concrete related surveillances after the fifth year following initial power operation may be adjusted based on the analysis of prior results.

Liner Integrity (SR 4.6.4.2)

Irradiation experiments on liner material specimens indicated that the material was capable of fulfilling its function throughout the design life of the plant. Approximately 750 specimens with dosimeters have been placed adjacent to the outside surface of the top head liner to permit detection of any shifts of the nil ductility transition temperature (NDT) characteristics of the 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 CYCLE and at specified times thereafter which is adequate to detect significant changes. Tensile specimens are not required for ASTM-E-185-70 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.

The interval for specimen removal and testing subsequent to the fifth refueling may be adjusted based on the analysis of prior results.

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 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 10% acceptance criteria is consistent with the acceptance criteria developed as part of the previous ISI procedures. The provisions are discussed in Section 5.13 of the FSAR. The SURVEILLANCE INTERVAL after the fifth year following initial power operation may be adjusted based on the analysis of prior results.

The interval specified for functional test and calibration of the instrumentation and alarms monitoring the core support floor and columns will ensure monitoring of any change in their structural integrity.

Penetration Well, and Isolation Valve Integrity (SR 4.6.4.3)

Integrity of PCRV penetration secondary pressure retaining boundaries is normally verified by continuous leakage monitoring and by periodic leakage testing of the penetration interspace per Specification 4.6.1.4. The specified examinations of accessible circumferential welds at structural discontinuities will ensure continued integrity of the secondary pressure boundary at these locations.

Isolation valve integrity is required in the unlikely event of a PCRV LCS tube rupture or a purification cooler tube rupture. A 72 hour Action is acceptable considering the redundancy in the associated systems.

Examination of accessible penetration closures, flow restrictors, and equipment restraint or support components ensures 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 interval specified for valve testing is adequate to ensure proper valve operation when isolation of the interspace auxiliary piping is required.

Verification of the proper operation of the interlocks associated with valves in the purification and regeneration piping of one train of the helium purification system is accomplished once per REFUELING CYCLE. Proper operation of these interlocks minimizes the potential likelihood of the maximum credible accident described in FSAR, Section 14.8. Proper operation of the interlocks is observed each time a purification train is regenerated or placed in service with the primary coolant system.

The interval specified for testing the helium purification cooler wells is adequate to verify the well integrity, as well as that of primary coolant boundary components located therein.

References:

(1) PSC letter, Warembourg to Johnson, dated March 5, 1985 (P-85071), Subject: Revised Tendon Surveillance Program.

(2) PSC letter, Warembourg to Johnson, dated March 18, 1985 (P-85084), Subject: PCRV Tendon Prestressing System, Transmitting attached report: "Tendon Surveillance Fort St. Vrain Nuclear Generating Station."

(3) GA Technologies document 907441/A, "Fort St. Vrain - PCRV Tendon Evaluation", April 30, 1984, provided as Attachment 7 to PSC letter, Warembourg to Berkow, dated 7/29/86 (P-86491).

(4) PSC letter, Warembourg to Calvo, dated 7/20/87 (P-87234), Subject: Fourth Six-Month PCRV Tendon Interim Surveillance Report and Proposal for Revised Interim Surveillance Program.

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PCRV AND CONFINEMENT SYSTEMS

3/4.6.5 REACTOR BUILDING CONFINEMENT

REACTOR BUILDING CONFINEMENT INTEGRITY

LIMITING CONDITION OF OPERATION

3.6.5.1 REACTOR BUILDING CONFINEMENT INTEGRITY shall be maintained.

APPLICABILITY: POWER, LOW POWER, STARTUP, SHUTDOWN*, AND REFUELING*

ACTION:

a. POWER, LOW POWER, and STARTUP:

Without REACTOR BUILDING CONFINEMENT INTEGRITY, restore integrity within 2 hours or be in SHUTDOWN within the subsequent 24 hours.

b. SHUTDOWN* and REFUELING*:

Without REACTOR BUILDING CONFINEMENT INTEGRITY, immediately suspend all operations involving CORE ALTERATIONS, control rod movements resulting in positive reactivity changes, or movement of IRRADIATED FUEL in the reactor building.

SURVEILLANCE REQUIREMENTS

4.6.5.1 REACTOR BUILDING CONFINEMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by;

1. Verifying that all reactor building overpressure protection system louvers are in the closed position, and

2. Verifying that all exterior doors and hatches are in the closed position except as permitted by Specification 4.6.5.1.b and 4.6.5.1.c below.

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* During CORE ALTERATIONS or handling of IRRADIATED FUEL in the reactor building.

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b. By verifying prior to each time that the truck doors to the truck bay or the personnel access door in the truck door are opened, that the truck bay floor hatch, the truck bay overhead sliding hatch, and the internal personnel door in the truck bay are closed.

c. By verifying prior to each time that the truck bay floor hatch, the truck bay overhead sliding hatch, and/or the internal personnel door are opened, that the truck doors and the personnel access door in the truck door are closed.

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BASIS FOR SPECIFICATION LCO 3.6.5.1/SR 4.6.5.1

The integrity of the reactor building confinement, in conjuction with operation of the ventilation exhaust system, limits 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 guidelines (FSAR Section 14.10.3.4).

However, the worst postulated primary coolant leak accident, Design Basis Accident #2 (DBA-2, FSAR Section 14.11), was analyzed with no credit taken f/ e reactor building exhaust fans and filters. In Section 14.11 it is shown that in the absence of these items the total duration site boundary doses would still be considerably less than the 10CFR100 limits. Credit has been taken for operation of the reactor building exhaust fans and filters in the analysis of the Maximum Credible Accident (MCA, FSAR Section 14.8) and the Design Basis Accident #1 (DBA-1, FSAR Section 14.10) to mitigate the consequences of these accidents. (FSAR Section 6.2.2).

With confinement integrity intact, the leak rate of the reactor building through other various and minor pathways is not a significant parameter with regard to limiting off-site doses, as is shown in FSAR Section 6.2.4.1.

Two hours is a reasonable time to reestablish reactor building confinement envelope integrity.

The integrity of the reactor building confinement is normally maintained with the exterior doors and the overpressure protection system louvers closed. These positions are verified on a monthly basis, consistent with plant isolation valve position verification.

The internal doors and hatches of the truck bay are closed to ensure integrity of the reactor building confinement prior to the opening of the truck door to the truck bay.

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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 two of the three exhaust trains OPERABLE, with each train consisting of one exhaust fan (C-7301, C-7302, or C-7302S) and an associated filter assembly (F-7301, F-7302, or F-7302S).

APPLICABILITY: POWER, LOW POWER, STARTUP, SHUTDOWN and REFUELING*

ACTION:

a. POWER, LOW POWER, and STARTUP

1. With reactor building internal pressure greater than or equal to atmospheric pressure, restore it to subatmospheric within 6 hours or be in at least SHUTDOWN within the next 24 hours.

2. With only one exhaust train OPERABLE, restore an inoperable train to OPERABLE status within 7 days or be in at least SHUTDOWN within the next 24 hours.

b. SHUTDOWN and REFUELING*

1. With the reactor building internal pressure greater than or equal to atmospheric pressure, immediately suspend all operations involving CORE ALTERATIONS, control rod movements resulting in positive reactivity changes, or movement of IRRADIATED FUEL in the reactor building.

2. With only one exhaust train OPERABLE, restore an inoperable train to OPERABLE status within 7 days, or suspend all operations involving CORE ALTERATIONS, control rod movements resulting in positive reactivity changes, or movement of IRRADIATED FUEL in the reactor building.

* During CORE ALTERATIONS or handling of IRRADIATED FUEL in the reactor building.

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.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following fire, chemical release, or painting with other than low solvent paints, 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 1% 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 between 17,100 and 23,000 cfm 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%, and

3. Verifying a flow rate between 17,100 and 23,000 cfm per train during system operation when tested in accordance with ANSI N510-1975.

c. 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%.

d. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches water gage while operating at a flow rate between 17,100 and 23,000 cfm for each filter train.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank 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 while operating the system at a flow rate between 17,100 and 23,000 cfm per train.

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber bank 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 between 17,100 and 23,000 cfm per train.

BASIS FOR SPECIFICATION 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 most accident conditions of operation. Additional information is provided in the BASIS for LCO 3.6.5.1/SR 4.6.5.1 and FSAR Section 6.2.4.

The system consists of three trains, two of which are normally in continuous operation (FSAR Section 6.2.3.2), with the third normally on standby. The design flow rate for each train is 19,000 cfm. Allowing 10%, the minimum flow rate is 17,100 cfm. Based on past performance data, a maximum flow rate of 23,000 cfm is also specified. One train is sufficient to maintain the reactor building subatmospheric and thereby minimize unfiltered fission product release from the building. With only one exhaust fan operating, the ventilation system controls will throttle fresh air supply to the air handler in order to reduce the pressure.

The reactor building is maintained in a subatmospheric condition to ensure that all air leakage will be inward and to minimize unfiltered fission product release from the building. The ventilation system was designed to maintain a subatmospheric condition of approximately 1/4 inch water gauge negative (FSAR 6.2.3.2). In actual practice, the reactor building pressure is normally indicated approximately 0.15 to 0.20 inches water gauge negative, depending on building activity and ventilation equipment configuration. There is an alarm at approximately 0.08 inches water gauge negative, and the air supply will fully close if the building pressure increases to atmospheric (FSAR Section 14.7.3).

Bypass leakage and penetration for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and dioctyl phthalate (DOP) respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. The filter/adsorber penetration and bypass acceptance limits in the surveillances are applicable based on a HEPA filter efficiency of 95% and charcoal adsorber efficiency of 90% assumed in the AEC staff's Safety Evaluation (Table 4.3, Safety Evaluation, Jan. 20, 1972; and FSAR Section 14.12.3).

The surveillance frequencies specified establish system performance capabilities.

The activated carbon adsorber in the affected unit will be replaced if a representative sample fails to pass the iodine removal efficiency test. Any HEPA filters found defective will be replaced.

If fire, chemical release, or painting, occurs such that the HEPA filter or charcoal adsorber could become significantly contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed, as required, for operational surveillance. Reactor building exhaust train(s) OPERABILITY should be verified per SR 4.6.5.2.b following:

1. Painting, except when water-base or equivalent paint is used,

2. Any spray (aerosol generating) painting (includes water-base or equivalent paint),

3. Fires that exceed 1 hour in duration, or

4. Any uncontrolled release/spillage of 5 gallons or more of any chemical material which could reasonably be expected to interfere with the charcoal to adsorb methyl iodide.

A pressure drop across the combined HEPA filter and charcoal adsorber of less than 6 inches of water gauge at the filter design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

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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, 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 at least 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 individually exercising each louver group through an open-closed cycle.

b. At least once per 18 months by:

1. Simulating an overpressurization signal and verifying that each louver group opens following switch actuation at a pressure signal of at least 3 inches water gauge differential pressure, and fully closes when the pressure signal drops below 2 inches water gauge differential pressure, and

2. Verifying that the pneumatic cylinders for operation of each louver group have at least 1800 psig in their nitrogen backup supply.

BASIS FOR SPECIFICATION LCO 3.6.5.3/SR 4.6.5.3

The purpose of the reactor building overpressure protection system is to maintain the integrity of the reactor building by relieving the pressure inside the building when it equals or exceeds atmospheric pressure by 3 inches water gauge, as detected in the turbine building. In the unlikely event of the occurrence of an increase of pressure inside the building exceeding 3 inches water gauge, the louvers would open, relieving the pressure, and then begin to automatically close at 2 inches water gauge, restoring the integrity of the reactor building (FSAR 6.2.3.4) and maintaining the potential doses from the occurrence to as low as practicable.

The louvers are opened by spring pressure, and automatically closed (or held closed) by air pressure acting through a pneumatic cylinder. The cylinders for each louver group will operate with 80 to 100 psig service air. A manually actuated nitrogen supply serves as a backup.

The OPERABILITY of 64 louver panels is required to prevent the pressure buildup in the reactor building from exceeding design limits. Each louver panel provides 12.02 sq. ft. of free flow area and the reactor building design pressure of 10 inches water gauge is not reached if the total louver free flow area is greater than 760 sq. ft. Requiring at least 70 louver panels to be operable provides a 10% margin of safety.

In an actual overpressure condition, once the pressure in the pneumatic cylinder is released, the building pressure acting on the louvers will assist the springs in quickly opening them. During surveillance testing, verifying that the louvers open assures that the pneumatic cylinders are released and that reactor building overpressure will be able to sufficiently open the louvers. Smooth louver operation is demonstrated during the quarterly exercise. Louvers may be tested collectively, individually, or in groups. The 18 month test verifies pressure switch setpoints and the circuitry of the system.

Plant administrative controls and Specification 3/4.6.5.1, Reactor Building Integrity, ensure that the quarterly stroke test of the louvers will be commensurate with overall plant safety and will minimize the potential for an unmonitored radioactive release path through the louvers.

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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 in any 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 OPERABLE.

APPLICABILITY: POWER, LOW POWER, STARTUP*, SHUTDOWN*, and REFUELING*

ACTION: With none of the above combinations OPERABLE, restore either of the above combinations to OPERABLE status within 72 hours or:

a. When in POWER, LOW POWER or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

b. When in SHUTDOWN or REFUELING, suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least once per REFUELING cycle, the boiler feed pumps shall be demonstrated OPERABLE by driving two helium circulators simultaneously, at an equivalent 8000 rpm (at atmospheric pressure), on the water turbines using the emergency feedwater header. This testing may be performed in conjunction with Specification 4.5.1.1.a.

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* With CALCULATED BULK CORE TEMPERATURE greater than 760 degrees F and with the PCRV pressurized to greater than 100 psia.

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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. The boiler feed pumps are not SAFE SHUTDOWN COOLING equipment. Feedwater is required only in the rapid depressurization accident (DBA-2) as discussed in FSAR Section 14.11, and in the maximum credible depressurization accident (MCA) as discussed in FSAR Section 14.4.3.2.

The requirement for both a motor driven and a turbine driven boiler feed pump provides redundancy in equipment and diversity in drive power source. Requiring a combination of two boiler feed pumps with a backup steam supply available from an auxiliary boiler if the two steam driven pumps are used provides additional redundant capability for shutdown cooling. Either auxiliary boiler will provide adequate steam supply for driving the turbine-driven boiler feed pumps if the motor-driven boiler feed pump is inoperable. Normal steam supply for the turbine-driven boiler feed pumps is provided via the cold reheat piping. The auxiliary boilers provide additional motive capability for these pumps in the event normal steam sources are unavailable. When the motor driven boiler feed pump is OPERABLE, sufficient diversity in drive power sources is ensured and the auxiliary boiler is not required. Analyses performed for High Energy Line Breaks demonstrate that the most limiting time for restart of forced circulation following DBA-2 is 60 minutes (FSAR section 14.11.2.2) and the requirement for an OPERABLE auxiliary boiler ensures that steam can be available for forced circulation drive in support of this 60-minute requirement. It is noted that the auxiliary boilers are not maintained in an operating condition at reactor power levels greater than 65%, to reduce their potential energy contribution in the event of a high energy line break.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of primary coolant helium flow.

The requirement for the boiler feed pumps to provide sufficient cooling after DBA-2 addresses a highly incredible event (FSAR Section 14.11.1) and a 72 hour ACTION time provides adequate margin should the boiler feed pumps become inoperable. Other means for cooling are available using condensate or boosted firewater coupled with physically redundant piping, valves and components.

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BASIS FOR SPECIFICATION LCO 3.7.1/SR 4.7.1 (Continued)

The boiler feed pumps will be demonstrated OPERABLE by driving two circulators simultaneously at an equivalent of 8000 rpm at the depressurized condition. This is the circulator flow assumed in FSAR Section 14.11.2 to assure fuel integrity in the highly unlikely event of DBA-2. The helium circulator flow requirements are discussed in the BASIS for Specification 3/4.5.1.

During plant operation, whenever the boiler feed pumps are operating, various support equipment, such as the condensate system will also be operating as required.

Since the boiler feed pumps are only relied upon in the event of a depressurization accident, they are not required if the reactor is already cooled and depressurized. Thus, the Applicability of this specification has been defined as whenever the PCRV is pressurized above 100 psia, and the CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

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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:

a. A water level in the steam/water dump tank (T-2201) less than or equal to 45 inches;

b. Both steam/water dump valves OPERABLE per loop (Loop 1: HV-2215 and HV-2217, Loop 2: HV-2216 and HV-2218);

c. The steam/water dump tank safety valves (V-2270 and V-2275) with a setpoint of 860 plus or minus 10 psig;

d. The steam/water dump tank block valves sealed open (Loop 1: V-2212 and V-2213, Loop 2: V-2242 and V-2243);

e. OPERABLE interlocks between Loop 1 and Loop 2 dump systems which prevent the simultaneous dumping of both loops; and

f. OPERABLE feedwater isolation/emergency feedwater block valves and feedwater control valves (Loop 1: HV-2201/HV-2203, and FV-2205, Loop 2: HV-2202/HV-2204, and FV-2206).

APPLICABILITY: POWER and LOW POWER

ACTION:

a. With steam/water dump tank level greater than 45 inches, restore to the required level within 8 hours or be in at least STARTUP within the next 24 hours.

b. With any one of the above components of the steam/water dump system inoperable, restore the system to OPERABLE status within 7 days, 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 92 days, by operating the steam/water dump valves through one complete cycle of full travel.

c. At least once per 18 months by:

1. Testing the electrical interlocks between Loop 1 and Loop 2, to verify interlock action preventing the simultaneous dumping of both loops.

2. Verifying the feedwater isolation/emergency feedwater block valves and feedwater control valves close on receiving actuation signal within the times specified below.

VALVE	TIME (Seconds)
HV-2201	1
HV-2202	1
HV-2203	10
HV-2204	10
FV-2205	2
FV-2206	2

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. As noted in FSAR Section 14.5 and Appendix A.12, the steam graphite reaction is strongly temperature dependent and insignificant below 900 degrees F. The average graphite temperature in STARTUP is less than 500 degrees F and therefore the system is not required for operation in STARTUP.

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 overfilling the tank and lifting the safety valves due to hydrostatically filling the tank during a dump. The tank level is verified once per 24 hours.

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. However, for single failure considerations, both valves are required to be OPERABLE in POWER and LOW POWER. 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 (RT-93250-12 and RT-93251-12) are covered in Specification 4.3.2.2.

The feedwater isolation valves (HV-2201 and HV-2202), emergency feedwater block valves (HV-2203 and HV-2204), 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 in Specification 4.3.1.

BASIS FOR SPECIFICATION LCO 3.7.1/SR 4.7.1 (Continued)

The only tests which are required to ensure OPERABILITY of the system are confirmation that the feedwater isolation valves and the dump valves are OPERABLE. The feedwater isolation valves are also required to function for SAFE SHUTDOWN COOLING. Each dump valve will be tested individually at 92 day intervals. Since the steam generators are not to be dumped during the test, which is performed during plant operation, the block valve is closed downstream of the dump valve being tested to isolate the dump tank from the secondary loop. After operation of the dump valve, the block valve is again locked open, returning the dump valve to service.

Testing of the interlocks that prevent simultaneous dumping of both loops is performed once every 18 months. The interval of 18 months for testing an installed electrical interlock is adequate assurance that a simultaneous two loop steam dump will not occur for a highly unlikely accident. Performance of a valve stroke test for valves actuated during a steam dump test at least once per 18 months with the specified acceptance criteria (time), provides assurance that the energy deposited to the steam dump tank is within design limits and that oxidation of the graphite in the core is within analyzed bounds. Thus, integrity of the core and the steam dump system will be maintained for a steam tube rupture event.

Operation during the first 11 years at Fort St. Vrain has resulted in pinhole tube leaks in the steam generators on two occasions, both of which were characterized by a slow (over several hours) entry of moisture into the PCRV with eventual reactor shutdown and manual loop dump. The design accident as analyzed in the FSAR, section 6.4, assumes a complete offset rupture of a steam generator subheader with a significant amount of moisture entering the PCRV in a short time. The time from rupture to the automatic inlet initiation of a dump is about 8.5 seconds at 100% power to 39.5 seconds at 25% load. Because the probability for this design accident is small, a restoration time for the steam/water dump system of 7 days is considered acceptable.

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 hot reheat steam line power operated relief valves (PCV-5221-1 and PCV-5221-2) shall be OPERABLE.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: With either hot reheat steam line power operated relief valve (PCV-5221-1 or PCV-5221-2) inoperable, restore the valve(s) to OPERABLE status within 72 hours or be in at least SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The hot reheat steam line power operated relief valves shall be demonstrated OPERABLE at least once per 12 months, or at the next scheduled plant shutdown if not tested during the previous 12 months, by exercising the valves.

BASIS FOR SPECIFICATION LCO 3.7.1.3/SR 4.7.1.3

The hot reheat steam line is equipped with two power operated pressure relief valves actuated by loss of voltage or degraded voltage on two of the three 480 VAC essential buses, 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 line relief valves discharge to atmosphere. Transients in which the hot reheat steam power operated relief valves are utilized, coincident loss of outside electric power and main turbine TRIP with a loss of one standby generator, are discussed in Sections 10.3.1 and 10.3.2 of the FSAR.

The ACTION to restore any inoperable power operated relief valve (PCV-5221-1 or PCV-5221-2) within 72 hours or be in SHUTDOWN within 24 hours is acceptable since there are six ASME Code spring loaded safety valves located downstream in that same reheat steam line.

Annual exercising of the hot reheat steam line power operated relief valves will ensure freedom of movement and proper function.

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PLANT AND SAFE SHUTDOWN COOLING SUPPORT

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 I-131 and 6.8 uCi/cc of tritium.

APPLICABILITY: POWER, LOW POWER, and STARTUP

ACTION: With the specific activity of the secondary coolant system greater than 0.009 uCi/cc of I-131 or 6.8 uCi/cc of tritium, be in at least LOW POWER within 12 hours and SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the above limits by analyzing for I-131, tritium, and gross beta concentration:

a. At least once per 24 hours, when the secondary coolant activity level exceeds 10% of the above limits.

b. At least once per 7 days, when the activity level is less than 10% of the above limits.

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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 orders of magnitude less than the 10 CFR 100 dose guideline values, in the event of an accident involving the loss of outside power, and main turbine TRIP, combined with failure of one diesel generator to start (FSAR Section 10.3.2). In this case, about 52,000 gallons of secondary coolant water could be vented to the atmosphere as steam if corrective action were not taken. Conservatively assuming a dilution factor of 2.7 E-3, and no partition factor of the iodine between the steam released and the water not released, a 2 hour exposure dose of only about 1.5 rem to the thyroid would be obtained. Using the same conservative assumptions for tritium, a 2 hour exposure dose of only about 0.5 rem to the whole body would be obtained. Therefore, an orderly (24 hour) shutdown is appropriate if a limit is exceeded.

This Specification is applicable only in POWER, LOW POWER, and STARTUP because these are the only times activity can be added to the secondary coolant. Furthermore, the accident referenced above is initiated from a condition in which the turbine is operating, which can occur only in POWER 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. More frequent monitoring of the secondary system is provided by the daily sampling and analysis frequency if the activity level exceeds 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.

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PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.7.1.5 The steam generator economizer-evaporator-superheater (EES) and reheater safety valves (V-2214, V-2215, V-2216, V-2245, V-2246, V-2247, V-2225 and V-2262) shall be OPERABLE with setpoints in accordance with Table 4.7.1-1.

APPLICABILITY: POWER*

ACTION:

a. With one of the required EES safety valves inoperable in either or both loops or with one reheater safety valve inoperable, restore the required valve to OPERABLE status within 72 hours or restrict plant operation as follows:

   1. With an EES safety valve inoperable, restrict plant operation to less than 50% of RATED THERMAL POWER.

   2. With a reheater safety valve inoperable, be in at least SHUTDOWN within the next 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 The superheater and reheater safety valves shall be demonstrated OPERABLE by testing in accordance with the applicable ASME Code requirements to verify setpoints. The test frequency is specified in the ASME Code, and the lift settings are specified in Table 4.7.1-1.

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* Setpoint verification is not required until 7 days after achieving steady state plant operating conditions at a power level above 50% RATED THERMAL POWER.

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TABLE 4.7.1-1

STEAM GENERATOR SAFETY VALVES

VALVE NUMBER	LIFT SETTINGS
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

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PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.1.6 a. At least one safety valve for each operating section of the steam generator shall be OPERABLE with its setpoint in accordance with Table 4.7.1-1.*

b. The provisions of Specification 3.0.4 are not applicable.

APPLICABILITY: LOW POWER, STARTUP, SHUTDOWN and REFUELING

ACTION: With less than the above required safety valves OPERABLE:

a. When in LOW POWER or STARTUP, restore the inoperable valve to OPERABLE status within 1 hour or be in at least SHUTDOWN within the next 24 hours, or

b. When in SHUTDOWN or REFUELING, restore the inoperable valve to OPERABLE status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F or suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.1.6 No additional surveillances required beyond those identified per Specification 4.7.1.5.

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* Setpoint verification requires power levels not included in the Applicability of this Specification. Where the test interval has been exceeded or the setting has been affected by valve maintenance, valve settings shall be estimated and verified per Specification 3.7.1.5.

BASIS FOR SPECIFICATIONS LCO 3.7.1.5/SR 4.7.1.5 AND LCO 3.7.1.6/SR 4.7.1.6

The economizer-evaporator-superheater (EES) section of each steam generator loop is protected by three spring-loaded safety valves, each with one-third nominal relieving capacity of each loop. The reheater section of each steam generator loop is protected from overpressure transients by a single safety valve. These steam generator safety valves are described in the FSAR, Section 10.2.5.3.

These steam generator safety valves are designed to relieve steam and can be damaged by rapid cyclic actuations that occur when they relieve water. To protect these valves, only one EES safety valve and the reheater safety valve are maintained in service in each loop, through startup evolutions until about 30% power. When the steam conditions support their use, all the steam safety valves are placed in service. The use of one safety valve per steam generator section at low power levels is in accordance with the requirements of the applicable piping code, as it is capable of relieving the available flow. Also, there are other power actuated valves that are capable of relieving pressure from the main steam and reheat piping.

The above valves are required to be tested in accordance with ASME Section XI, IGV requirements every 5 years (or less, depending on failures) or after maintenance. To satisfy the testing criteria, the valves must be tested with steam. Since these valves are permanently installed in steam piping, the appropriate means for testing requires the plant to be operating at steady state conditions, and close to the steam conditions expected at the setpoint. Power levels above 50% RATED THERMAL POWER are sufficient to achieve this. Also, 7 days ensures setpoint verification within a reasonable time, noting that the test schedules are such that all valves are not tested at the same time and thus, some valves will normally be OPERABLE.

During all MODES, with one EES safety valve inoperable, plant operation is restricted to a condition for which the remaining safety valves have sufficient relieving capability to prevent overpressurization of any steam generator section. Conversely, with any reheater safety valve inoperable, plant operation is restricted to a more restrictive MODE.

A 72-hour action time for repair or SHUTDOWN due to inoperable safety valves ensures that these valves are returned to service in a relatively short period of time, during which an overpressure transient is unlikely. Operation at power for 72 hours does not result in a significant loss of safety function for any extended period.

The setpoints for the safety valves identified in Table 4.7.1-1 are those values identified in the FSAR with tolerances applied such that the Technical Specifications incorporate an upper bound setpoint. This is consistent with not incorporating normal operating limits in these Specifications.

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PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

CONDENSATE PUMPS

LIMITING CONDITION FOR OPERATION

3.7.1.7 At least one of the two 12 1/2% condensate pumps (P-3106 or P-3106S) shall be OPERABLE.

APPLICABILITY: POWER and LOW POWER

ACTION: With no OPERABLE 12 1/2% condensate pump, restore at least one pump to OPERABLE status within 72 hours or be in at least STARTUP within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.7 The 12 1/2% condensate pumps shall be demonstrated OPERABLE at least once per 3 months (92 days) by performance of a functional test.

BASIS FOR SPECIFICATION LCO 3.7.1.7/SR 4.7.1.7

The condensate pumps are used to assure forced circulation cooling under certain normal and emergency conditions. Condensate is supplied via the emergency condensate header to the helium circulator water turbines and to the steam generator EES or reheater sections. The condensate pumps are not SAFE SHUTDOWN COOLING equipment.

There are two 60% condensate pumps and two 12 1/2% condensate pumps. The 12 1/2% condensate pumps are addressed in this specification because they are capable of being powered from an essential power bus and would be capable of supplying emergency condensate requirement in the event of a loss of off-site power. (FSAR Section 10.3.2) A single pump is specified because the condensate pumps are a part of the design of FSV, but they are not ultimately relied upon for forced cooling, per the FSAR. The SAFE SHUTDOWN COOLING equipment at FSV relies upon the firewater pumps (see Specification 3/4.5.5) to assure forced cooling (FSAR Sections 14.4.2, 10.3.9).

72 hours is a reasonable period of time to restore an inoperable pump to operable status, consistent with repair times allowed for SAFE SHUTDOWN COOLING equipment.

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 each group of valves maintained greater than 2500 psig,

c. At least two OPERABLE hydraulic pumps, and

d. Hydraulic oil reservoir temperature less than 150 degrees F.

APPLICABILITY: POWER, 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 secondary coolant loop within 1 hour, and be in SHUTDOWN within 24 hours.

b. With the hydraulic oil temperature exceeding 150 degrees F, restore the oil temperature to within its limit within 24 hours or be in SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.2 The hydraulic power system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying:

1. That hydraulic fluid pressure to each group of valves is greater than 2500 psig, and

2. That hydraulic oil reservoir temperature is less than 150 degrees F.

b. At least once per 18 months by verifying that the standby pump automatically starts when system pressure drops to less than 2800 psig.

BASIS FOR SPECIFICATION LCO 3.7.2/SR 4.7.2

Each secondary coolant loop hydraulic power system is designed with the following: 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. (FSAR Section 9.11).

The hydraulic oil temperature limit of 150 degrees F corresponds to the system design temperature and minimizes oxidation of the hydraulic fluid, thereby enhancing its service life. The temperature of the hydraulic oil reservoirs, which are immediately downstream of the hydraulic oil coolers, is alarmed in the control room.

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. A 1 hour ACTION time is provided to isolate the affected loop, in an effort to regain the capability to supply at least 2500 psig and/or the OPERABILITY of at least one accumulator for the affected valve group. If these efforts are not successful, reactor shutdown is required within the next 24 hours.

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 inoperable, the ability to totally isolate the affected coolant loop is ensured by the selective grouping of valves.

In the event of loss of all hydraulic power in one system, all flow pressure and 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.

BASIS FOR SPECIFICATION LCO 3.7.2/SR 4.7.2 (Continued)

Verifying that the standby hydraulic pump starts when system pressure falls below 2800 psig ensures that a system pressure of at least 2800 psig will be maintained at the pumps; this will provide at least 2500 psig at the valve actuators, taking into consideration line losses. This surveillance will be performed once per 18 months.

Verifying that hydraulic fluid pressure is greater than 2500 psig once per 24 hours ensures that the minimum pressure required to operate the hydraulic valves in the secondary coolant system is available.

Verifying that the hydraulic oil temperature is less than 150 degrees F once per 24 hours ensures that system oil temperature is within design limits.

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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, 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 SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 Each instrument air system shall be demonstrated OPERABLE:

a. At least once per 24 hours, by verifying that the instrument air receiver is pressurized to greater than or equal to 85 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.

BASIS FOR SPECIFICATION LCO 3.7.3/SR 4.7.3

The instrument air system ensures that sufficient air is available for continued operation of essential instrumentation, control devices, the reserve shutdown system, and pneumatic power operated valves required for SAFE SHUTDOWN COOLING (as discussed in Section 10.3.9 of the FSAR).

The instrument air system is a normally operating system. The system is demonstrated OPERABLE by ensuring that receiver pressure is maintained greater than 85 psig. The pressure at which the third standby instrument air compressor starts is 85 psig and this ensures adequate air pressure for operating essential valves and instrumentation. The description of this system is presented in FSAR Section 9.9.

Either instrument air system is capable of supplying air to safe shutdown components for purposes of SAFE SHUTDOWN COOLING. A 72-hour ACTION time limits the interval for which redundant instrument air systems are unavailable to a short period of time.

The SURVEILLANCE INTERVAL for verifying system pressure, 24 hours, is sufficient to assure that instrument air is available to operate components required for SAFE SHUTDOWN COOLING. An 18 month SURVEILLANCE INTERVAL for a functional test of the instrument air system verifies all automatic controls, alarms and components are OPERABLE.

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.4 SERVICE WATER SYSTEM-OPERATING

LIMITING CONDITION FOR OPERATION

3.7.4.1 The service water system shall be OPERABLE with:

a. At least two of three service water pumps (P-4201, P-4202, or P-4202S) OPERABLE,

b. An OPERABLE service water flow path to SAFE SHUTDOWN COOLING users which are required to be OPERABLE (emergency diesel coolers, instrument air compressors and after coolers, and the reactor plant cooling water/PCRV liner cooling heat exchangers: E-4601, E-4602, E-4603, and E-4604), and

c. An OPERABLE flow path from the circulating water makeup system to the service water pump pit.

APPLICABILITY: POWER, LOW POWER, STARTUP*, SHUTDOWN*, and REFUELING*

ACTION:

a. With only one service water pump OPERABLE, restore at least two pumps to OPERABLE status within 72 hours, or

1. When in POWER, LOW POWER, or STARTUP, be in at least SHUTDOWN within the next 24 hours, or

2. When in SHUTDOWN, suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes, or movement of IRRADIATED FUEL.

* With the CALCULATED BULK CORE TEMPERATURE greater than 760 degrees F.

SPECIFICATION 3.7.4.1 (Continued)

b. With an inoperable service water flow path to either the emergency diesel coolers, or the instrument air compressors and after coolers:

1. Declare the affected component(s) inoperable and restore the flow path to OPERABLE status within 72 hours, or

2. Initiate backup cooling from the firewater system with at least two OPERABLE firewater pumps within 1 hour and restore service water to the affected component(s) within 72 hours of initial loss of service water, or

3. Be in at least SHUTDOWN within the following 24 hours.

c. With an inoperable service water flow path to the reactor plant cooling water/PCRV liner cooling system heat exchangers:

1. Declare the affected component inoperable, and restore the flow path to OPERABLE status within 72 hours, or

2. Verify backup cooling capability with at least two OPERABLE firewater pumps within 1 hour and restore service water to the affected component within 72 hours of initial loss of service water, or

3. Be in at least SHUTDOWN within the next 24 hours.

d. With backup cooling initiated or the capability verified as in ACTION c.2 above and with only one firewater pump OPERABLE, restore two firewater pumps to OPERABLE status within 1 hour, or be in SHUTDOWN within the next 24 hours.

e. With an inoperable flow path from the circulating water makeup system to the service water pump pit, restore the flow path to OPERABLE status within 24 hours, or

1. When in POWER, LOW POWER, or STARTUP, be in SHUTDOWN within the next 24 hours, or

2. When in SHUTDOWN, suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes, or movement of IRRADIATED FUEL.

SURVEILLANCE REQUIREMENTS

4.7.4.1 The service water system shall be demonstrated OPERABLE:

a. At least once per 24 hours, by verifying that the above required service water pumps are OPERABLE and circulating service water.

b. At least once per 31 days by:

1. Verifying that each valve in the flow path from the circulating water makeup pump discharge to the service water pump pit that is not locked, sealed, or otherwise secured in place, is in its correct position, and

2. Functionally testing each service water pump.

c. At least once per 12 months by verifying the performance and mechanical condition of each service water pump. This may be performed at the next scheduled plant shutdown, if not performed during the previous 12 months.

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.4 SERVICE WATER SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.4.2 The service water system shall be OPERABLE with:

a. One service water pump (P-4201, P-4202, or P-4202S) OPERABLE,

b. An OPERABLE service water flow path to SAFE SHUTDOWN COOLING users (emergency diesel coolers, instrument air compressor and after coolers, and the reactor plant cooling water/PCRV liner cooling heat exchangers: E-4601, E-4602, E-4603, and E-4604), and

c. An OPERABLE flow path from the circulating water makeup system to the service water pump pit.

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING*

ACTION: With any of the requirements identified in 3.7.4.2 a, b, or c above inoperable,

a. Provide alternate cooling capability to the required SAFE SHUTDOWN COOLING users, or

b. Be in at least SHUTDOWN within 12 hours and restore the equipment to OPERABLE status prior to reaching a CALCULATED BULK CORE TEMPERATURE of 760 degrees F, or suspend all operations involving CORE ALTERATIONS or control rod movements resulting in positive reactivity changes.

* With the CALCULATED BULK CORE TEMPERATURE less than or equal to 760 degrees F.

SURVEILLANCE REQUIREMENTS

4.7.4.2 The service water system shall be demonstrated OPERABLE:

a. At least once per 24 hours, by verifying that the above required service water pump is OPERABLE and circulating service water.

b. At least once per 31 days by:

1. Verifying that each valve in the flow path from the circulating water makeup pump discharge to the service water pump pit that is not locked, sealed, or otherwise secured in place, is in its correct position, and

2. Functionally testing each service water pump.

c. At least once per 12 months by verifying the performance and mechanical condition of each service water pump. This may be performed at the next scheduled plant shutdown, if not performed during the previous 12 months.

BASIS FOR SPECIFICATIONS LCO 3.7.4.1 and 3.7.4.2

Approximately 25% of the 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 in LOW POWER or POWER, there will normally be at least two service water pumps in OPERATION. The requirement for two OPERABLE service water pumps is for single failure considerations during SAFE SHUTDOWN COOLING. The firewater system serves as a backup to the service water system. The fire water system OPERABILITY requirements are given in Specification 3.5.4. 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 firewater system provides an independent source of cooling water for the service water system for all safe shutdown essential water requirements. Firewater is a backup to the Reactor Plant Cooling Water System as a source of once-through cooling water to the liner cooling tubes. The firewater system is applicable during all MODES of operation.

With the CALCULATED BULK CORE TEMPERATURE less than or equal to 760 degrees F, the circulating water system may also be used as an alternate to service water. This provides an alternate during service water system maintenance outages that does not present the tube fouling problems associated with firewater.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV interval components regardless of the amount, including total absence, or reversal, of primary coolant helium flow.

During POWER, LOW POWER, STARTUP, SHUTDOWN, and REFUELING, with the CALCULATED BULK CORE TEMPERATURE greater than 760 degrees F, with only one service water pump OPERABLE, a restoration time of 72 hours is provided to restore another pump to OPERABLE status. During this 72 hours, service water needs can be met by the redundant pump.

BASIS FOR SPECIFICATIONS LCO 3.7.4 and 3.7.4 (Continued)

With an inoperable flow path to the emergency diesel coolers or instrument air compressors and after coolers, initiation of backup cooling is required, because the affected component may be automatically initiated to perform SAFE SHUTDOWN COOLING. With an inoperable flow path to the reactor plant cooling water/PCRV liner cooling heat exchangers, backup cooling capability must be verified, so that it can be manually initiated if required. Actual initiation of backup cooling to the PCRV LCS heat exchangers is undesirable except in an actual emergency to minimize possibility of tube fouling. 72 hours is provided to restore the flow path to OPERABLE status. This is consistent with the restoration times provided for in the system OPERABILITY requirements for the above components. If the flow path cannot be restored within 72 hours, backup cooling will be initiated (firewater) or the capability verified within 1 hour to restore the equipment to OPERABLE status. Both firewater pumps must be OPERABLE. The surveillance requirements for firewater pump OPERABILITY are given in Specification 3.5.4. If neither of these conditions can be met, the plant must be in SHUTDOWN within 24 hours. If only one firewater pump is OPERABLE, a second pump must be restored to OPERABLE status within 1 hour, or the plant must be in SHUTDOWN within 24 hours. The 1 hour restoration time is required because service water is unavailable, and the firewater supply has no redundant capability.

With the flow path from the circulating water makeup system to the service water pump pit inoperable, a restoration time of 24 hours is provided to restore the flow path to OPERABLE status. This ACTION time is adequate, since backup cooling (firewater) can be initiated, and considering the makeup requirements of the service water system.

During STARTUP, SHUTDOWN, and REFUELING, with the CALCULATED BULK CORE TEMPERATURE less than or equal to 760 degrees F, a service water pump, flow path to SAFE SHUTDOWN COOLING users of service water, and a flow path from the circulating water makeup system to the service water pump pit are required to be OPERABLE prior to the time calculated to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. This ACTION ensures that the plant will remain in a stable condition when service water is unavailable due to maintenance, testing, or unanticipated outages.

The service water system is normally operating, including the required OPERABLE flow paths, which demonstrates system OPERABILITY. The flow path to the standby diesel generators is demonstrated OPERABLE by the surveillance testing of the diesel generators.

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PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.5 PRIMARY COOLANT DEPRESSURIZATION

LIMITING CONDITION FOR OPERATION

3.7.5 a. Two flow paths for primary coolant depressurization shall be OPERABLE, each from the primary coolant system through a helium purification train to the reactor building ventilation system exhaust.

b. At least 650 gallons of liquid nitrogen shall be maintained in the liquid nitrogen storage tank (T-2501).

APPLICABILITY: POWER, LOW POWER, STARTUP, SHUTDOWN*, and REFUELING*

ACTION:

a. With only one of the above required helium purification train depressurization flow paths OPERABLE due to regeneration of the second purification train,

1. Initiate action to regenerate the second helium purification train within 24 hours of its removal from service and restore it to OPERABLE status within the following 31 days, and

2. Be in at least SHUTDOWN within 72 hours after failure to regenerate within 31 days.

3. The provisions of Specification 3.0.4 are not applicable.

b. With only one of the above required helium purification train depressurization flow paths OPERABLE other than due to regeneration, restore two purification train flow paths to OPERABLE status within 7 days, or be in at least SHUTDOWN within the next 24 hours.

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* With the CALCULATED BULK CORE TEMPERATURE greater than 760 degrees F.

SPECIFICATION 3.7.5 (Continued)

c. With none of the required helium purification train depressurization flow paths OPERABLE:

   1. Restore at least one train to OPERABLE status within 12 hours or be in at least SHUTDOWN within the next 24 hours, and

   2. Restore at least two purification train depressurization flow paths to OPERABLE status in accordance with ACTION a or b, as applicable.

d. With less than 650 gallons of liquid nitrogen in the nitrogen storage tank, restore the liquid nitrogen storage inventory to 650 gallons within 24 hours or be in at least SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The helium purification train depressurization flow path(s) shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the liquid nitrogen storage tank (T-2501) contains at least 650 gallons of liquid nitrogen.

b. At least once per 31 days by functionally testing each purification cooling water pump (P-4701 and P-4702).

c. At least once per 12 months by verifying the performance of each purification cooling water pump. This may be performed at the next scheduled plant shutdown, if not performed during the pervious 12 months.

d. At least once per 18 months by cycling (through one complete cycle of full travel) the valves for routing helium gas to the reactor building ventilation exhaust and for cooling the high temperature filter adsorber (HTFA).

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 an extended Loss Of Forced Circulation (LOFC), one train of the helium purification system must be OPERABLE to depressurize the primary coolant system.

Depressurization is required to reduce the density of the primary coolant and thereby reduce the heat transfer to the PCRV liner cooling system. This action, combined with adjustments in the liner cooling system flow distribution and pressure ensures that the integrity of the PCRV liner is maintained (FSAR Appendix D.1.1.1, D.1.2.1.2, Section 14.10.3.1)

The normal depressurization flow path provides for a filtered release through the following OPERABLE components: the high temperature filter adsorber (HTFA), the helium purification cooler, the helium purification dryer, the low temperature gas-to-gas heat exchanger, the Low Temperature Adsorber (LTA), purified helium filter and associated piping and valves leading to the reactor building exhaust.

In emergency conditions, a depressurization flow path may be established through a regeneration train by bypassing block valve interlocks. This is not a normal flow path, but is an acceptable alternate under emergency conditions. That is, it is the same path as the normal primary coolant depressurization path with the LTA and/or purification system dryer bypassed. The LTA would be bypassed because it is cooled by liquid nitrogen normally and the flow path may be restricted due to freezing. With the LTA bypassed, the regeneration train can be effectively used for depressurization of the PCRV and the consequences are still well below 10CFR100 limits, e.g. bounded by the accidents described in FSAR Section 14.11.2.8.

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 during an LOFC accident. This is acceptable for a limited period of time due to the availability of an alternate depressurization flow path via the regeneration piping.

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BASIS FOR SPECIFICATION LCO 3.7.5/SR 4.7.5 (Continued)

A total of 650 gallons of liquid nitrogen is required to provide refrigeration for the low temperature adsorber during depressurization (FSAR Section 9.6.6).

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.

The HTFA coolers are used only in the event of an extended LOFC accident. The coolers are dry during normal operation and are isolated from the Reactor Plant Cooling Water system by two valves with a tell-tale drain. The cycling of these valves is performed in a manner that does not introduce water into the coolers, as any residual water would remain there due to the U-tube design.

PLANT AND SAFE SHUTDOWN COOLING SUPPORT

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, and
6. The main and unit auxiliary transformers.

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,
8. The turbine building side of the "G" wall for the congested electrical cable area, and
9. The reactor building side of the "J" wall for the congested electrical cable area.

SPECIFICATION 3.7.6.1 (Continued)

c. Manually operated fixed spray systems

1. The 480 volt switchgear room, and

2. 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 and 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 and 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 the requirements of ACTIONS a.2 or a.3 can be justified, they may be initiated at that time.

c. The provisions of Specifications 3.0.3 and 3.0.4 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.

SPECIFICATION 4.7.6.1 (Continued)

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

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

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.

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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 capability 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, if safety-related equipment is present in the affected area.

Verifying valve position, valve actuation, inspecting the headers and nozzles, and verifying that each header is unobstructed ensures OPERABILITY should a fire occur.

A continuous fire watch shall be performed once per 20 minutes.

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 a continuous fire watch with backup fire suppression equipment for the affected emergency diesel generator room(s).

b. The provisions of Specifications 3.0.3 and 3.0.4 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 that the carbon dioxide storage tank level is greater than 30% and that the pressure is greater than 285 psig,

b. 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

c. 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."

BASIS FOR SPECIFICATION LCO 3.7.6.2/SR 4.7.6.2

The OPERABILITY of the carbon dioxide fire 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, a continuous fire watch will be established within 1 hour and the affected emergency diesel generator room(s) will be protected by a carbon dioxide hose station or portable extinguishers, which will provide an equivalent level of fire protection for postulated fires.

Verifying level and pressure in the carbon dioxide storage tank once per 7 days ensures that a sufficient amount of carbon dioxide is available to suppress any postulated fires in the emergency diesel generator rooms (as discussed in FSAR Section 9.12.2.3).

The monthly valve position check includes valves CO14, CO19, CO31, and CO32 and verifies a clear flow path for CO2 delivery as needed.

Verifying valve and damper actuation by a simulated actuation signal ensures a discharge of the proper carbon dioxide concentration (34%) and a soaking time of 30 minutes (as discussed in FSAR Section 9.12.2.3).

Performing a "puff test" will ensure that the distribution headers are not blocked.

A continuous fire watch shall be satisfied by performance of a check at least once per 20 minutes.

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 equipment room,

c. 480 volt switchgear room,

d. Building 10 - switchgear room and ground level,

e. Building 10 - ground level under mezzazine floor, and

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 1 hour establish a continuous fire watch with backup fire suppression equipment for the affected room(s).

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.3 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying pressure in the Halon bottles.

b. At least once per quarter by verifying that the Halon storage tank quantity is at least 95% of full charge and pressure is at least 90% of full charge pressure for the following systems:

1. Building 10 - switchgear room and ground level,

2. Building 10 - ground level under mezzanine floor, and

3. Building 10 - battery room

c. At least once per 6 months, by verifying that the Halon storage tank weight is at least 95% of full charge weight (or level) and pressure is at least 90% of full charge pressure for the following systems:

1. Control room,

2. Auxiliary electric equipment room, and

3. 480 volt switchgear room

d. 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.

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 system 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 equipment room, 3) the 480 volt 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.

Halon is supplied from full capacity main cylinders and 100% spare reserve cylinders; either the main or the reserve cylinders may be used to satisfy the specification requirements. The OPERABILITY of the associated Heating Ventilating and Air Conditioning (HVAC) isolation dampers ensures 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 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 equipment room.

The surveillance requirements ensure that the minimum OPERABILITY requirements of the Halon suppression systems are met. A semi-annual surveillance ensures that a sufficient volume of Halon is in the storage tanks by verifying either the weight or the level of the tanks. For the building 10 Halon system, storage tank pressure and quantity are verified quarterly, because the storage tank weight cannot be verified without removing the tanks. Quantity is determined by use of a surveillance method which is acceptable to the American Nuclear Insurers (ANI), such as the heat tape and gun method. The 31-day surveillance for checking pressure in the Halon systems is adequate verification of proper valve lineup as a mispositioned valve results in discharge of the Halon. 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.

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 listed 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. If the hose station is the primary means of fire suppression, within 1 hour either:

   1. Ensure that the nearest OPERABLE hose station can provide coverage for the area normally protected by the inoperable hose station, or

   2. Route additional equivalent capacity hose from the nearest OPERABLE hose station, to provide coverage for the area left unprotected by the inoperable hose station.

b. If the hose station is not the primary means of fire suppression, within 24 hours either:

   1. Ensure that the nearest OPERABLE hose station can provide coverage for the area normally protected by the inoperable hose station, or

   2. Route additional equivalent capacity hose from the nearest OPERABLE hose station, to provide coverage for the area left unprotected by the inoperable hose station.

c. Perform an evaluation to determine whether the physical routing of the fire hose (per ACTION a.2 or b.2) would result in a recognizable hazard to operating personnel, plant equipment, or the hose itself, in which case the fire hose will be stored at the outlet of the OPERABLE hose station.

d. The provisions of Specifications 3.0.3 and 3.0.4 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 ensure 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 greater than or equal to 175 psig.

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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	4904
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

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

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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

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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.

A fire hose station shall be considered a primary means of fire suppression if there are no Halon, CO2, or spray or sprinkler systems providing coverage to that area.

Visually inspecting the fire hose stations, removing the hose and reracking it, replacing any degraded gaskets in the couplings, verifying flow, and hydrostatically testing the hose will ensure OPERABILITY should a fire occur.

Fire hose stations may be added without prior License Amendment to Table 3.7.6-1, provided a revision to Table 3.7.6-1 is included with a subsequent License Amendment request.

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, and

e. Number 11 southeast of the turbine building.

APPLICABILITY: At all times

ACTION: With any of the above listed yard fire hydrants or associated hydrant hose houses inoperable:

a. Within 1 hour, locate additional equivalent capacity hose in an adjacent OPERABLE hydrant hose house to provide coverage for the area left unprotected by the inoperable fire hydrant or associated hydrant hose house, if it is the primary means of fire suppression, or

b. Within 24 hours, locate additional equivalent capacity hose in an adjacent OPERABLE hydrant hose house to provide coverage for the area left unprotected by the inoperable fire hydrant or associated hydrant hose house, if it is not the primary means of fire suppression.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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SURVEILLANCE REQUIREMENTS

4.7.6.5 Each of the above yard fire hydrants and associated hydrant hose houses listed in Specification 3.7.6.5 above shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the hydrant hose house to ensure all required equipment is at the hose se,

b. At least once per 184 days (once during March, April or May and once during September, October or November), 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 greater than or equal to 175 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.

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, additional lengths of equivalent capacity hose will be located in an adjacent OPERABLE hydrant hose house within 1 hour if the fire hydrant or associated hydrant hose house is a primary means of fire suppression, or within 24 hours if it is not a primary means of fire suppression. This will ensure that any area left unprotected by an inoperable fire hydrant or associated hydrant hose house will be protected should a fire occur. A fire hydrant or associated hydrant hose house shall be considered the primary means of fire suppression if there are no spray or sprinkler systems providing coverage to that area.

Visually inspecting the hydrant hose house, the barrel, and the hydrant, replacing any degraded gaskets in the couplings, verifying flow and hydrostatically testing the hose will ensure OPERABILITY should a fire occur.

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.7 FIRE RATED BARRIERS

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:

1. Establish a continuous fire watch on at least one side of the affected barrier, or

2. Verify the OPERABILITY of fire detectors on at least one side of the inoperable barrier and establish an hourly fire watch patrol, or

3. Perform an evaluation and determine that the areas on both sides of the inoperable barrier do not contain equipment which is required to be OPERABLE.

b. The requirements of ACTION a.3 may be performed subsequent to the establishment of a fire watch and implemented at any time.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7 The above required fire barriers and penetration sealing devices shall be demonstrated OPERABLE:

a. At least once per 24 hours, by verifying that each unlocked fire door is closed.

b. At least once per 7 days, by verifying the position of each locked fire door.

c. At least once per 18 months by:

1. Performing a visual inspection of the exposed surfaces of:

a) Each fire barrier,

b) Each fire damper and associated hardware, and

c) 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.

2. Performing a functional test of the automatic fire dampers.

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 an OPERABLE condition.

BASIS FOR SPECIFICATION LCO 3.7.7/SR 4.7.7

The OPERABILITY of the fire barriers and barrier penetrations ensures 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. A safety-related fire area is any area of the plant which contains safety-related equipment, as listed in Tables 1.4-1 and 1.4-2 of the FSAR.

In the event that a fire barrier does not remain intact, a continuous fire watch (once every twenty minutes) 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.

Verifying the position of each unlocked and locked fire door once per 24 hours and 7 days respectively, ensures that the fire doors are in their as-designed condition to confine or retard fires from spreading to adjacent portions of the facility. Visually inspecting the exposed surface of each fire barrier, damper and associated hardware, and 10% of each type of sealed penetration once per 18 months, and functionally testing each automatic fire damper once per 18 months ensures that the release, closing mechanism and latches will be able to perform their design function when required. Verifying that the seal is returned to its original condition following any maintenance or repair work which disturbs the fire retardant material in the sealed penetrations, ensures that it was not damaged or altered in any way to prevent it from performing its design function.

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.8 STEAM LINE RUPTURE DETECTION/ISOLATION SYSTEM VALVES

LIMITING CONDITION FOR OPERATION

3.7.8 All valves actuated by the Steam Line Rupture Detection/Isolation System (SLRDIS) shall be OPERABLE.

APPLICABILITY: POWER, LOW POWER, and STARTUP (above 2% RATED THERMAL POWER)

ACTION:

a. With any one valve actuated by SLRDIS inoperable, restore the valve to OPERABLE status within 72 hours or reduce power to below 2% within the next 12 hours.

b. With two or more valves actuated by SLRDIS inoperable, restore all but one valve to OPERABLE status within 24 hours or reduce power to below 2% within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.8 The valves actuated by SLRDIS shall be demonstrated OPERABLE:

a. At least once per 92 days, by cycling each testable valve through at least one complete cycle.

b. At least once per REFUELING CYCLE, by cycling each valve not testable during plant operation through at least one complete cycle.

BASIS FOR LCO 3.7.8/SR 4.7.8

As in the Steam Line Rupture Detection/Isolation System (SLRDIS) instrumentation Specification 3/4.3.1, SLRDIS valves are only requireu above 2% RATED THERMAL POWER. An inoperable valve or associated equipment is allowed for 72 hours. High energy line break analysis for environmental qualification assumes the worst-case single active failure. Thus, a single valve inoperable for up to 72 hours is within the bounds of analysis. When two or more valves and/or associated equipment is inoperable, 24 hours is allowed to restore the inoperable equipment. Repairs may be performed while the plant is at power, thus, minimizing thermal cycling of plant and installed equipment. SLRDIS, including valve functions and stroke times, is described in FSAR Section 7.3.10.

The surveillances exercise the valves to assure proper movement when required. Testable valves are those that can be exercised without impacting the operation of their associated system.

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 emergency ventilation system shall be OPERABLE in the recirculation mode with minimum makeup with:

a. Both the control room emergency filter fan (C-7506) and the control room supply fan (C-7504X) OPERABLE, and

b. The control room emergency makeup ventilation filter (F-7502) OPERABLE.

APPLICABILITY: At all times

ACTION: POWER, LOW POWER and STARTUP

a. With one of the above required fans inoperable, but with control room positive pressure greater than or equal to 0.05 inches water gauge, restore the inoperable fan to OPERABLE status within 7 days or be in SHUTDOWN within the next 24 hours.

b. With one of the above required fans inoperable, and control room positive pressure less than 0.05 inches water gauge, restore the inoperable fan to OPERABLE status within 24 hours or be in SHUTDOWN within the next 24 hours.

c. With the control room emergency makeup ventilation filter (F-7502) inoperable restore the filter to OPERABLE status within 72 hours or be in SHUTDOWN within the next 24 hours.

SHUTDOWN and REFUELING

With the above requirements for the control room emergency ventilation system not met, restore the system to OPERABLE status within 7 days or suspend all operations involving CORE ALTERATIONS, control rod movements resulting in positive reactivity changes, or movement of IRRADIATED FUEL.

SURVEILLANCE REQUIREMENTS

4.7.9 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days, by initiating, from the control room, flow through the High Efficiency Particulate Air (HEPA) filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours.

b. 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 in any ventilation zone communicating with the system by:

1. Verifying that the ventilation system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% 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 system flow rate is greater than or equal to 450 ACFM.

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 3% at 30 degrees C, 95% RH.

3. Verifying a system flow rate of greater than or equal to 450 ACFM during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 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 3% at 30 degrees C, 95% RH.

SPECIFICATION 4.7.9 (Continued)

d. At least once per 18 months by:

1. Verifying that the pressure drop across the HEPA filters and charcoal adsorbers is less than 6 inches of water while operating the system at a flow rate of greater than or equal to 450 ACFM.

2. Verifying that on a simulated signal indicating high radiation in the reactor building ventilation exhaust, the system automatically switches into the recirculation mode with minimum makeup.

3. Verifying that the control room emergency ventilation system maintains the control room at a positive pressure in the following configurations:

a) With both the emergency filter fan and the control room supply fan OPERABLE, maintain at least 0.125 inches water gauge pressure.

b) With either the emergency filter fan or the control room supply fan inoperable, maintain at least 0.05 inches water gauge pressure.

e. 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% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate greater than or equal to 450 ACFM.

f. 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% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of greater than or equal to 450 ACFM.

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. FSAR Table 7.1-1 specifies that a slight positive pressure is assumed in the analysis of an operating environment. The recirculation mode with minimum makeup, also known as the hi-radiation mode or the minimum makeup mode, isolates the normal makeup and uses the normal control room supply and return fans to recirculate air through the air handling units. 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.

A 72-hour ACTION time associated with an inoperable control room emergency makeup ventilation filter (F-7502) is acceptable based on the fact that it is in service only when the control room ventilation system is in the minimum makeup mode. During normal control room ventilation, air is drawn from outside and bypasses the filter. In the event that the filter (F-7502) is inoperable and the control room ventilation system is in the minimum makeup mode, a backup source of air is provided to operators, if required, via the breathing air system.

The control room pressure can be maintained at a positive pressure of at least 0.125 inches water gauge in the recirculation mode with emergency makeup, with both the emergency filter fan and the supply fan operating. If either of these fans becomes inoperable, control room pressure can still be maintained positive, by shutting off the control room return fan and closing the toilet exhaust damper, in which case a positive pressure of at least 0.05 inches water gauge is provided. The channel accuracy for measuring control room pressure is 2% or less of the instrument range which is 2 inches. Thus, 0.04 inches water gauge may be assumed for total channel accuracy. Specifying a control room pressure of at least 0.05 inches water gauge as the surveillance requirement ensures positive pressure in the control room even in the event of a failure of one of the two required fans in the emergency ventilation line-up. Thus, the FSAR assumptions are verified through required surveillances. (FSAR Section 11.2.2 and Appendix C, Criterion 11).

BASIS FOR SPECIFICATION LCO 3.7.9/SR 4.7.9 (Continued)

The assumptions relative to control room positive pressure assume the access door to the control room is closed. However, due to plant security reasons, the access door may be opened and personnel access controlled by a full-time guard. In the event of an emergency, the door will be closed as required by plant operating procedures to ensure that a positive control room pressure is maintained.

Ensuring 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 slight positive pressure in the control room have been calculated to be 450 ACFM. That is, 450 ACFM, is enough to ensure that a positive reading on the control room pressure gauge exceeds the maximum channel accuracy.

The addition of these surveillance requirements is in response to NUREG-0737, Item III D.3.4.

PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 All Class I and Ia snubbers shall be OPERABLE.

APPLICABILITY: POWER 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 Class I and Ia snubber shall be demonstrated OPERABLE by performance of the following inservice inspection program in addition to the requirements of Specification 4.0.5.

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

Visual inspections shall be performed in accordance with the following schedule:

SPECIFICATION 4.7.10 (Continued)

No. of Inoperable Snubbers of Each Type per Specification Period	Subsequent Visual Inspection Period* #
0	18 months plus or minus 25%
1	12 months plus or minus 25%
2	6 months plus or minus 25%
3, 4	124 days plus or minus 25%
5, 6, 7	62 days plus or minus 25%
8 or more	31 days plus or minus 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 4.0.2 are not applicable.

SPECIFICATION 4.7.10 (Continued)

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 5% 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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SPECIFICATION 4.7.10 (Continued)

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;

SPECIFICATION 4.7.10 (Continued)

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.

SPECIFICATION 4.7.10 (Continued)

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.

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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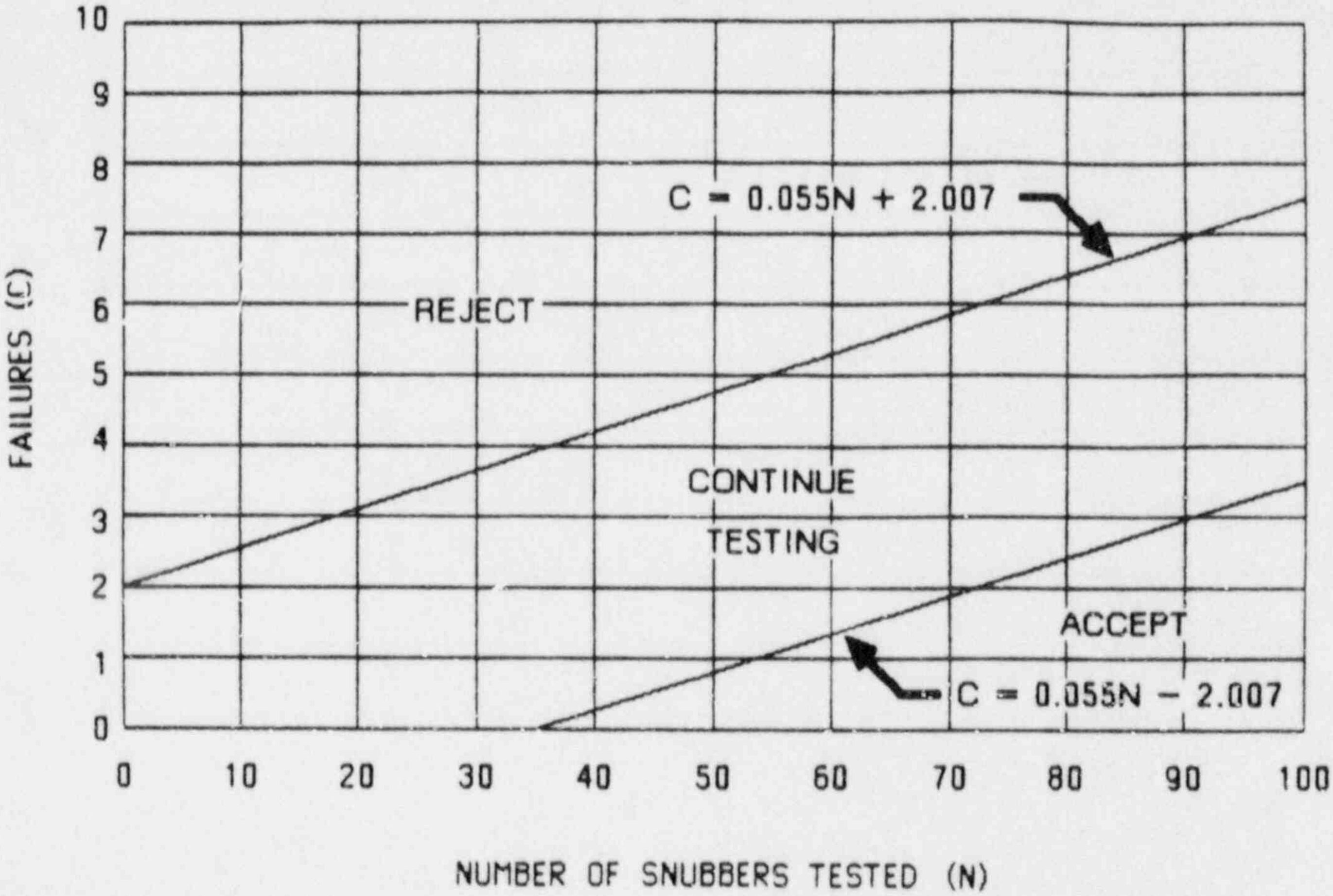

SAMPLE PLAN FOR NUMBER OF SNUBBERS TESTED

Figure 4.7.10-1

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BASIS FOR SPECIFICATION LCO 3.7.10/SR 4.7.10

All safety-related snubbers are required to be OPERABLE to ensure that the structural integrity of Class I systems is maintained during and following a seismic or other event initiating dynamic loads.

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 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 operation in POWER and LOW POWER.

Because snubber protection is required only during relatively low probability events, a period of 72 hours is provided for repair or replacement.

Plant operation at power levels up to 5% RATED THERMAL POWER is permitted without OPERABLE snubbers. PSC has performed an analysis which concludes that at 8% RATED THERMAL POWER, the plant can sustain a Loss of Forced Circulation, with the PCRV either pressurized or depressurized, with no PCRV liner cooling, and the fuel temperature does not approach 2900 degrees F and therefore no fuel failure or fission product releases would occur. (FSAR Section D.4.2). All plant systems that include snubbers could fail at 5% power and the consequences would be bounded by this analysis.

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%) may not be used to lengthen the required inspection interval. Any inspection where results require a shorter inspection interval will override the previous schedule.

BASIS FOR SPECIFICATION LCO 3.7.10/SR 4.7.10 (Continued)

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 solely via functional testing.

To ensure snubber functional reliability one of two functional testing methods is used with the stated acceptance criteria:

a. Functionally test 10% of a type of snubber with an additional 5% tested for each functional testing failure, or

b. 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 AC POWER SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following AC electrical power sources shall be OPERABLE:

a. The Unit Auxiliary Transformer (UAT) and the Main Power Transformer (MPT), with or without the generator output links installed, and the Reserve Auxiliary Transformer (RAT).

b. Two separate and independent Standby Diesel Generators (SDGs) with:

1. Each diesel fuel oil day tank containing a minimum of 325 gallons of fuel,

2. A minimum of 20,000 gallons of diesel fuel in underground storage including at least 5,500 gallons of diesel fuel in the diesel fuel oil storage tank (T-9201) and an OPERABLE flow path(s) capable of transferring fuel oil from storage to each day tank,

3. An OPERABLE water-jacket heater for each SDG diesel engine, and

4. Lubricating oil storage containing a minimum total volume of 100 gallons of lubricating oil.

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

* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

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SPECIFICATION LCO 3.8.1.1 (Continued)

ACTION:

a. With the UAT, the MPT, or the RAT of the above required off-site AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining off-site A.C. electrical power source by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either Standby Diesel Generator (SDG) has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirements 4.8.1.1.2.a.5 and 4.8.1.1.2.a.7 for each such SDG, separately, within 24 hours. Restore the inoperable off-site source to OPERABLE status within 24 hours, or be in at least SHUTDOWN within the next 24 hours.

b. With either SDG inoperable, demonstrate the OPERABILITY of the above required AC off-site sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour, and at least once per 8 hours thereafter. If the SDG became inoperable due to any cause other than preplanned preventive maintenance or testing and the remaining SDG has not been demonstrated OPERABLE in the previous 8 hours demonstrate the OPERABILITY of the remaining SDG by performing Surveillance Requirements 4.8.1.1.2.a.5 and 4.8.1.1.2.a.7, initiating the surveillances within 1 hour*. Restore the inoperable SDG to OPERABLE status within 72 hours or be in at least SHUTDOWN within the next 24 hours.

c. With the UAT, the MPT, or the RAT, and one SDG of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining off-site A.C. source by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour, and at least once per 8 hours thereafter. If the SDG became inoperable due to any cause other than preplanned preventive maintenance or testing and the remaining SDG has not been demonstrated OPERABLE within the previous 8 hours, demonstrate its OPERABILITY by performing Surveillance Requirements 4.8.1.1.2.a.5 and 4.8.1.1.2.a.7 initiating the surveillances within 1 hour*. Restore at least one of the inoperable AC electrical power sources to OPERABLE status within 12 hours, or be in at least SHUTDOWN within the next 24 hours. Restore the UAT, the MPT and the RAT to OPERABLE status within 24 hours and both SDGs to OPERABLE status within 72 hours from the time of initial loss of OPERABILITY, or be in at least SHUTDOWN within the next 24 hours.

* This test is required to be completed regardless of when the inoperable SDG is restored to OPERABLE.

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SPECIFICATION LCO 3.8.1.1 (Continued)

d. With one SDG inoperable, unless declared inoperable due to surveillance testing, perform the following in addition to ACTION b or c: ensure within 2 hours that all the required SAFE SHUTDOWN COOLING systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE SDG as a source are also OPERABLE, or be in at least SHUTDOWN within the next 24 hours.

e. With both SDGs inoperable, demonstrate the OPERABILITY of two off-site sources of the above required AC electrical power sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour, and at least once per 8 hours thereafter. Restore at least one of the inoperable SDGs to OPERABLE status within 2 hours, or be in at least SHUTDOWN within the next 24 hours. Restore both SDGs to OPERABLE status within 72 hours from the time of initial loss of OPERABILITY, or be in at least SHUTDOWN within the next 24 hours.

f. If results from a fuel oil sample taken per Surveillance Requirement 4.8.1.1.2.b.2 or 4.8.1.1.2.c are unacceptable, demonstrate the OPERABILITY of both SDGs, separately, by performing Surveillance Requirements 4.8.1.1.2.a.5 and 4.8.1.1.2.a.7 within 8 hours, if not performed within the last 7 days. Resample the fuel oil per Surveillance Requirement 4.8.1.1.2.b.2 or 4.8.1.1.2.c, as applicable, within 8 hours of receipt of the unacceptable result(s). If results from the second sample are unacceptable, the SDGs shall be declared inoperable. Drain the fuel oil from the contaminated tank and replace it with fresh fuel oil. Sample the fresh fuel oil per Surveillance Requirement 4.8.1.1.2.c.

g. With the diesel fuel oil storage tank (T-9201) unavailable, operation may continue provided the diesel fuel oil in underground storage tanks 1A and 1B has been demonstrated acceptable by performing Surveillance Requirements 4.8.4.d and 4.8.1.1.2.c and by performing Surveillance Requirement 4.8.1.1.2.c prior to addition of any fuel oil to these tanks. If the results from a fuel oil sample taken per Surveillance Requirement 4.8.4.d or 4.8.1.1.2.c are unacceptable, ACTION statement 3.8.1.1.f applies.

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SURVEILLANCE REQUIREMENTS

4.8.1.1.1 The required off-site sources of AC electric power shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. At least once per 18 months, during SHUTDOWN:

1. By verifying automatic transfer of house power supply from the Unit Auxiliary Transformer to the Reserve Auxiliary Transformer, and

2. By verifying that the Unit Auxiliary Transformer generator links can be tagged out and removed within 6 hours.

4.8.1.1.2 Each Standby Diesel Generator (SDG) 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 each diesel fuel oil day tank is at least 325 gallons,

2. Verifying the total fuel oil quantity in underground storage is at least 20,000 gallons with at least 5,500 gallons in the diesel fuel oil storage tank (T-9201) in addition to the day tanks,

3. Verifying the capability to transfer fuel oil from the underground storage system to the diesel fuel 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 100 degrees F,

5. Verifying the SDG diesel engines start from the normal pre-heated condition and accelerate to normal operating speed; the SDG voltage and frequency shall be 480 plus or minus 48 volts, and 60 plus or minus 1.2 Hz,

6. Verifying the lubrication oil inventory in storage is at least 100 gallons,

SPECIFICATION 4.8.1.1.2 (Continued)

7. Verifying the SDG is synchronized, loaded to 1150 KW plus or minus 50 KW* with two diesel engines per SDG and operates for at least 60 minutes,

8. Verifying the SDG is aligned to provide standby power to the associated essential buses, and

9. Verifying both air start receivers of the SDG are pressurized to greater than or equal to 120 psig.

b. At least once per 31 days:

1. By checking for and removing accumulated water from the fuel oil storage tanks: T-8401, T-8402, and T-9201, and

2. By obtaining a sample of fuel oil from the diesel fuel oil storage tank (T-9201) in accordance with ASTM D2276-78 and verifying that the total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, method A.

c. By sampling fuel oil in accordance with ASTM-D4057 prior to addition to the diesel fuel oil storage tank (T-9201) and:

1. By verifying a clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82 prior to addition to the diesel fuel oil storage tank (T-9201), and

2. By verifying within 14 days of obtaining the sample, when tested in accordance with ASTM-D975-77.

a) A flash point equal to or greater than 125 degrees F, and

b) 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.

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* This is a steady state load

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SPECIFICATION 4.8.1.1.2 (Continued)

d. At least once per 31 days by performing a CHANNEL FUNCTIONAL TEST of the SDG engine exhaust temperature "shutdown" and "declutch" function.

e. At least once per 18 months, during SHUTDOWN by:

1. Subjecting the SDG diesel engines to an inspection in accordance with the procedures prepared in conjunction with the manufacturer's recommendations.

2. Performing a CHANNEL CALIBRATION of the SDG "shutdown" and "declutch" engine protective functions.

3. Verifying the SDG capability to reject a load of greater than or equal to 202 KW while maintaining voltage at 480 plus or minus 48 volts and frequency at 60 plus or minus 1.2 Hz.

4. Verifying the SDG capability to reject a load of 1150 KW plus or minus 50 KW without tripping the SDG; the SDG voltage shall not exceed 552 volts during and following the load rejection.

5. Simulating an undervoltage relay actuation signal:

a) Verifying de-energization of the essential 480 VAC buses and load shedding from the essential 480 VAC buses.

b) Verifying the SDG diesel engines start on the auto-start signal, energize the essential 480 VAC buses within 60 seconds, start the auto-sequenced loads through the load sequencer, and OPERATE for greater than or equal to 5 minutes while the associated SDG 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, and

c) Verifying the overload and antimotoring SDG trip functions are bypassed when the SDGs are in the auto-start mode.

d) Verify that the load sequence timer is OPERABLE with the complete sequence loaded within plus or minus 10% of its design time.

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SPECIFICATION 4.8.1.1.2 (Continued)

6. Verifying that the auto-sequenced loads to each SDG do not exceed 1210 KW, with both SDG diesel engines operating.

7. Verifying the SDG's capability to:

    a) Synchronize with the off-site power source while the SDG is loaded with its emergency loads upon a simulated restoration of off-site power,

    b) Transfer its loads to the off-site power source, and

    c) Be restored to standby status.

8. Verifying the SDG operates for at least 24 hours; during the first 2 hours of this test, the SDG shall be loaded to 1260 KW plus or minus 60 KW*, and during the remaining 22 hours of this test, the SDG shall be loaded to 1150 KW plus or minus 50 KW*; the SDG voltage and frequency shall be 480 plus or minus 48 volts and 60 plus or minus 1.2 Hz; within 5 minutes after completing the 24 hour test, perform Surveillance Requirement 4.8.1.1.2.a.5.

f. At least once per 10 years, or after any modifications which could affect SDG interdependence, by starting both SDGs simultaneously, during SHUTDOWN, and verifying that both SDGs accelerate to normal operating speed.

g. At least once per 10 years by draining each underground storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or equivalent.

h. Reports - Standby Diesel Generator (SDG) 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.

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* This is a steady state load

TABLE 4.8.1-1

STANDBY DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests*	Test Frequency
$\leq$ 1	At least once per 31 days
$\geq$ 2	At least once per 7 days **
$\geq$ 3	See Table 4.8.1-2

* 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 Standby Diesel Generator (SDG) basis.

** This test frequency shall be maintained until 7 consecutive failure-free demands have been performed and the number of failures in the last 20 demands has been reduced to 1 or less.

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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 SDG diesel engine reliability improvement program implemented at the site. Minimum requirements for the report are indicated in Attachment 1 to this table.
5	11	Declare the SDG inoperable and perform a requalification test program for the affected SDG 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 S G reliability.

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ATTACHMENT 1 TO TABLE 4.8.1-2

REPORTING REQUIREMENT

As a minimum, the reliability improvement program report for a Commission 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. An analysis of failures and determination of root causes of failures.

C. An evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of On-site Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant.

D. An identification of all actions taken, or to be taken, to correct the root causes of failures defined in B above and to achieve a general improvement of SDG 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 SDG 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 as long as the affected SDG diesel engine 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 SDG since the last report for that SDG. 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 SDG improvement program and the schedule for implementation of those changes.

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ATTACHMENT 2 TO TABLE 4.8.1-2

STANDBY DIESEL GENERATOR REQUALIFICATION PROGRAM

A. Perform 7 consecutive successful demands, as specified in Surveillance Requirement 4.8.1.1.2.a, without a failure within 30 days of the SDG being restored to OPERABLE status, and 14 consecutive successful demands without a failure within 75 days of the SDG being restored to OPERABLE status.

B. If a failure occurs during the first 7 tests in the requalification test program, perform 7 successful demands without an additional failure within 30 days of the SDG being restored to OPERABLE status and 14 consecutive successful demands without a failure within 75 days of the SDG being restored to OPERABLE status.

C. If a failure occurs during the second 7 tests (tests 8 through 14) of A., above, perform 14 consecutive demands without an additional failure within 75 days of the failure.

D. If a second failure occurs during the requalification test program, be in at least SHUTDOWN within 24 hours.

E. During requalification testing, the SDG should not be tested more frequently than at 24-hour intervals.

F. After a SDG has been successfully requalified, subsequent repeated requalification tests will not be required for that SDG under the following conditions:

1. The number of failures in the last 20 valid demands is less than 5.

2. The number of failures in the last 100 valid demands is less than 11.

3. In the event that following successful requalification of a SDG, the number of failures is still in excess of the remedial action criteria (1. and/or 2. above) the following exception will be allowed until the SDG is no longer in violation of the remedial action criteria (1. and/or 2. 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 SDG is no longer in violation of the remedial action criteria above, the provisions of those criteria alone will prevail.

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AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.1 AC POWER SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following AC electrical power sources shall be OPERABLE:

a. Either the Reserve Auxiliary Transformer (RAT) or the Unit Auxiliary Transformer (UAT) and Main Power Transformer (MPT); and

b. One Standby Diesel Generator (SDG) with:

1. The diesel fuel oil day tank containing a minimum 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 fuel oil from storage to the day tank,

3. An OPERABLE water-jacket heater for each SDG diesel engine, and

4. Lubricating oil storage containing a minimum total volume of 50 gallons of lubricating oil.

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING

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* Whenever CALCULATED BULK CORE TEMPERATURE is less than or equal to 760 degrees F.

SPECIFICATION LCO 3.8.1.2 (Continued)

ACTION: With less than the above minimum required AC electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, control rod movement resulting in positive reactivity changes, or movement of IRRADIATED FUEL. Initiate corrective actions to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 No additional surveillance requirements are required other than those surveillances identified per Surveillance Requirements 4.8.1.1.1 and 4.8.1.1.2.

BASIS FOR SPECIFICATION LCO 3.8.1/SR 4.8.1

The OPERABILITY of the AC electrical power sources during POWER, LOW POWER, STARTUP and SHUTDOWN (whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F) 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 AC electrical power sources are adequate to satisfy the basis of General Plant Design Criteria No. 24 and No. 39 as stated in Appendix C of the FSAR.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of PRIMARY COOLANT FLOW.

The means of providing AC electrical power to plant auxiliaries during normal operation is through the Unit Auxiliary Transformer (UAT) energized by the main turbine-generator. The UAT can also be energized from the high voltage transmission lines through the station switchyard via the Main Power Transformer (MPT), after the generator links have been removed to isolate the main turbine generator. The UAT is connected to the 4160 VAC Buses 1 and 3, which are connected to the essential 480 VAC Buses 1 and 3, respectively.

Off-site AC electrical power during STARTUP, SHUTDOWN or loss of the UAT or MPT is supplied through the Reserve Auxiliary Transformer (RAT). The RAT is energized from the high voltage transmission lines via the station switchyard. The RAT is connected to the 4160 VAC Bus 2, which connects to essential 480 VAC Bus 2. Upon loss-of-power from the UAT, power supply to the plant auxiliaries is automatically transferred to the RAT.

On-site AC electrical power is supplied by two Standby Diesel Generators (SDGs), either of which has the capability to power all electrical auxiliaries that are essential for SAFE SHUTDOWN COOLING. Each SDG supplies essential 480 VAC Bus 1 or Bus 3, with the first-in SDG also energizing essential 480 VAC Bus 2.

A diesel fuel supply of 16,150 gallons is adequate to provide for operation of one SDG for one week, under required load conditions to shut down the plant and to maintain it in a safe condition. This reserve capacity provides ample time for obtaining additional fuel from off-site sources.

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BASIS FOR SPECIFICATION LCO 3.8.1/4.8.1 (Continued)

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 is based upon maintaining at least one of the redundant sets of on-site AC and DC 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.

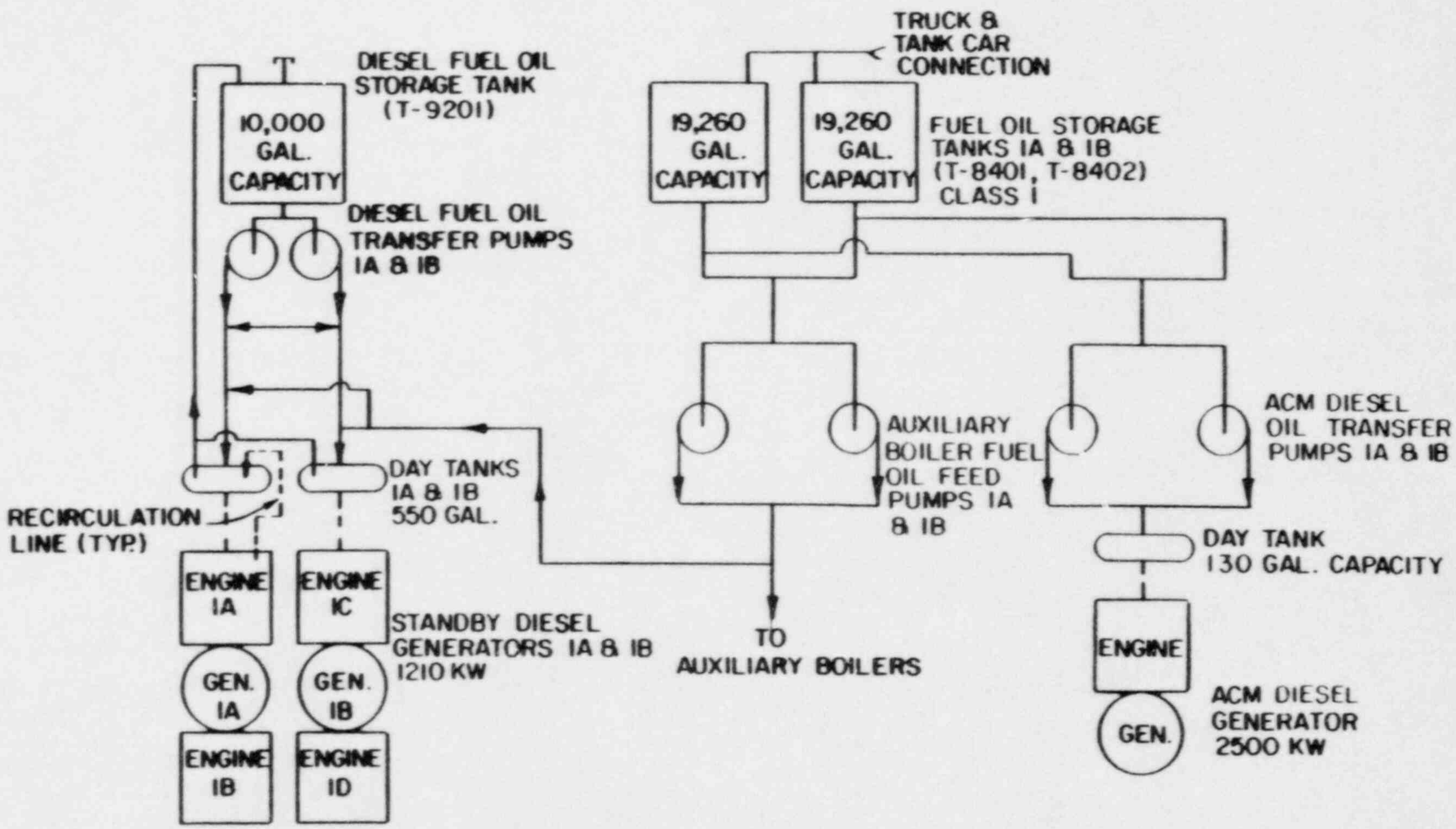
The term "verify" as used in the ACTION statements means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. The term "ensure" as used in ACTION statement 3.8.1.1.d allows 2 hours to verify OPERABLE or to restore to OPERABLE status affected equipment, with any additional ACTION not required, if in compliance.

The surveillance requirements are adequate to demonstrate the OPERABILITY of the off-site and on-site AC electrical power sources, such that their intended safety functions under postulated abnormal and accident conditions can be performed.

In particular, the surveillance requirements for the SDGs 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".

The SDGs are required to reach rated speed, voltage and frequency on demand. If an SDG does not reach these parameters or if the SDG fails to start due to depletion of the starting air receivers, the SDG start is considered a failure.

The SDG fuel oil sampling requirements are sufficient to assess fuel oil quality at Fort St. Vrain. With over 10 years of diesel generator operational experience, there have been no fuel oil related failures of the SDGs. Fuel oil is distributed between a diesel fuel oil storage tank for the SDGs and a shared tank arrangement with the Auxiliary Boiler. The turnover of diesel fuel in the underground storage tanks during SHUTDOWN, STARTUP and LOW POWER; the performance of Surveillance Requirements 4.8.4.d; and the performance of Surveillance Requirements 4.8.1.1.2.b and 4.8.1.1.2.c demonstrate the quality of diesel fuel oil in underground storage. Figure 3.8.1-1, Diesel Fuel Oil Systems, shows the tank and related piping arrangements.



DIESEL FUEL OIL SYSTEM

Figure 3.8.1-1

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AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.2 DC POWER SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following independent DC electrical power sources shall be OPERABLE:

a. Battery no. 1A and 1 dedicated battery charger,

b. Battery no. 1B and 1 dedicated battery charger, and

c. Battery no. 1C and 1 dedicated battery charger.

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

ACTION:

a. With one of the required batteries inoperable, verify the associated DC load is energized with a dedicated battery charger within 2 hours. If the battery is inoperable due to any cause other than an equalizing charge being performed, restore the inoperable battery to OPERABLE status within 24 hours or be in at least SHUTDOWN within the next 24 hours. If the battery is inoperable due to an equalizing charge being performed, restore the battery to OPERABLE status within 5 days or be in at least SHUTDOWN within the next 24 hours.**

b. With one dedicated battery charger inoperable, restore the inoperable dedicated battery charger to OPERABLE status within 24 hours or be in at least SHUTDOWN within the next 24 hours.

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* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

** A battery can be disconnected from the distribution system for up to 5 consecutive days and 10 cumulative days during any 92-day period, if an equalizing charge is being performed.

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SURVEILLANCE REQUIREMENTS

4.8.2.1 Each required DC electrical power source shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The Category A limits in Table 4.8.2-1 are met, and

2. The total battery terminal voltage is greater than or equal to 120.4 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 Category B limits in Table 4.8.2-1 are met,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than or equal to 150 micro-ohms, and

3. The average electrolyte temperature of at least 20% of the cells is 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,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material, and

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150 micro-ohms.

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SPECIFICATION SR 4.8.2.1 (Continued)

4. The battery charger will supply at least:

   a) 325 (+0, -25) amperes for at least 4 hours for battery chargers 1A and 1B,

   b) 200 (+0, -25) amperes for at least 4 hours for battery charger 1C, and

   c) 370 (+0, -25) amperes for at least 4 hours for battery charger 1D.

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 service discharge test.

e. At least once per 60 months, during SHUTDOWN, by:

   1. A performance discharge test on batteries 1A and 1B, separately, (at an average discharge rate of 310 amperes) over a period of 4 hours, or until the average battery terminal voltage reaches 1.81 volts/cell; the test shall be acceptable if, after 3.2 hours the battery is capable of producing at least 310 amperes, and the average battery terminal voltage is greater than 1.81 volts/cell.

   2. A performance discharge test on battery 1C (at an average discharge rate of 183 amperes) over a period of 4 hours, or until the average battery terminal voltage reaches 1.81 volts/cell; the test shall be acceptable if, after 3.2 hours the battery is capable of producing at least 183 amperes and the average battery terminal voltage is greater than 1.81 volts/cell.

   Once per 60-month interval, the performance discharge test may be performed in lieu of the battery service discharge test in Surveillance Requirement 4.8.2.1.d.

f. At least once per 18 months, during SHUTDOWN, by a performance discharge test of battery capacity 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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TABLE 4.8.2-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A (1)	CATEGORY B (2)	
	Pilot Cell	Connected Cell(s)	
Parameter	Limits	Limits	Allowable value (3)
Electrolyte Level	Greater than minimum level indication mark, and less than one quarter inch above maximum level indication mark.	Greater than minimum level indication mark, and less than one quarter inch above maximum level indication mark.	Above top of plates, and not overflowing.
Float Voltage	≥ 2.13 volts	≥ 2.13 volts (4)	≥ 2.07 volts
Specific Gravity (5)	≥ 1.205 (6)	≥ 1.195 for each connected cell.	Each connected cell not more than .020 below the average of all connected cells.
		Average of all connected cells > 1.205.	Average of all connected cells > 1.195. (6)

'( )' refer to notes on following page.

Notes for Table 4.8.2-1:

(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 the 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 7 days.

(3) If any Category B parameter is not within its allowable value, declare the battery inoperable.

(4) Measured cell voltages of cells warmer than average may be corrected for electrolyte temperature, if measured cell voltage is less than the acceptance criteria.

(5) Corrected for electrolyte temperature and level.

(6) Or battery charging current is less than 2 amperes when on float charge.

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AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.2 DC POWER SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following independent DC electrical power sources shall be OPERABLE:

a. Battery no. 1A and 1 dedicated battery charger, or

b. Battery no. 1B and 1 dedicated battery charger, or

c. Battery no. 1C and 1 dedicated battery charger.

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING

ACTION: With only one of the above batteries or only one dedicated battery charger OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, control rod movement resulting in positive reactivity changes, or movement of IRRADIATED FUEL. Initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The required DC electrical power sources shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

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* Whenever CALCULATED BULK CORE TEMPERATURE is less than or equal to 760 degrees F.

BASIS FOR SPECIFICATION LCO 3.8.2/SR 4.8.2

The OPERABILITY of the DC electrical power sources during POWER, LOW POWER, STARTUP, and SHUTDOWN (whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F) ensures that sufficient power will be available, as required to perform the intended safety functions under postulated abnormal and accident conditions. Batteries 1A and 1B are each adequate to supply the required safe shutdown DC loads for not less than 4 hours, while battery 1C supplies a DC power source for one channel of the PPS loads, following the loss of all AC power. The batteries provide the source of power, through the inverters and static transfer switches to the AC instrument power buses. The minimum specified requirements for independent and redundant DC electrical power sources are adequate to satisfy the basis of General Plant Design Criteria No. 24 and No. 39 as stated in Appendix C of the FSAR.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of PRIMARY COOLANT FLOW.

The FSV DC electrical power system consists of three batteries and four battery chargers. Two batteries, 1A and 1B, are connected to DC buses, 1A and 1B, respectively, and the third battery, 1C, is connected directly to inverter/static transfer switch, 1C. Three battery chargers, 1A, 1B, and 1C, energized from the essential 480 VAC buses, maintain batteries 1A, 1B, and 1C, respectively, on float charge during normal operation. The fourth battery charger, 1D, is utilized to supply bus 1A, or bus 1B or inverter 1C, through bus ties, while maintaining independence between the other DC power sources and the DC loads. The use of either battery chargers, 1A, 1B, or 1C or backup battery charger, 1D to supply the associated DC load, constitutes a dedicated battery charger.

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 AC and DC 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.

BASIS FOR SPECIFICATION LCO 3.8.2/SR 4.8.2 (Continued)

In ACTION 3.8.2.1.d a battery can be inoperable for 5 consecutive days and 10 cumulative days during a 92 day period, if an equalizing charge is being performed. An equalizing charge per the manufacturer requires a nominal 3 to 5 days to complete and is required every 2 to 3 months. At FSV a battery/battery charger is disconnected from the distribution system when an equalizing charge is being performed to preclude damage to the DC loads. Due to the fact that a battery/battery charger can readily be placed back in service by reconnecting it to the system, this configuration is not considered a significant degradation of the DC power sources or the on-site distribution system.

The surveillance requirement for demonstrating the OPERABILITY of the 1A, 1B and 1C batteries is based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1987, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service discharge and performance discharge tests ensures the effectiveness of the charging system, the ability of the battery to handle high discharge rates, and the adequacy of the battery capacity with respect to the rated capacity and emergency load requirements.

Table 4.8.2-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than or equal to 2.13 volts and 0.010 below the manufacturer's full charge specific gravity is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than or equal to 2.13 volts and not more than 0.020 below the full charge specific gravity of 1.215 with an average specific gravity of all the connected cells not more than 0.010 below the full charge specific gravity of 1.215, ensures the OPERABILITY and capability of the battery.

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BASIS FOR SPECIFICATION LCO 3.8.2/SR 4.8.2 (Continued)

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2-1 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the recommended full charge specific gravity, ensures that the decreases in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, not more than 0.040 below the full charge specific gravity, ensures that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.3 ON-SITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical buses shall be energized in the specified manner with tie breakers open between redundant buses:

a. Essential 480 VAC Bus 1 energized from the 4160 VAC Bus 1,**

b. Essential 480 VAC Bus 2 energized from the 4160 VAC Bus 2,**

c. Essential 480 VAC Bus 3 energized from the 4160 VAC Bus 3,**

d. 120 VAC non-interruptible Bus 1A/1A-1 energized through their respective inverter/static transfer switch from the preferred power source, 125 VDC Bus 1A,

e. 120 VAC non-interruptible Bus 1B/1B-1 energized through their respective inverter/static transfer switch from the preferred power source, 125 VDC Bus 1B,

f. 120 VAC non-interruptible Bus 1C/1C-1 energized through their respective inverter/static transfer switch from the preferred power source battery No. 1C, and 1 dedicated battery charger,

g. 125 VDC Bus 1A energized from associated battery No. 1A, and 1 dedicated battery charger, and

h. 125 VDC Bus 1B energized from associated battery No. 1B, and 1 dedicated battery charger.

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

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* Whenever CALCULATED BULK CORE TEMPERATURE is greater than 760 degrees F.

** The requirement for the tie breakers to be open between redundant buses is not applicable with respect to the 4160 VAC buses.

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SPECIFICATION FOR 3.8.3.1 (Continued)

ACTION:

a. With one of the required 480 VAC buses not energized in the required manner, reenergize the bus in the required manner within 8 hours or be in at least SHUTDOWN within the next 24 hours.

b. With one 120 VAC non-interruptible bus not energized in the required manner from its preferred source*:

1. Reenergize the 120 VAC non-interruptible bus from the essential 480 VAC bus backup source via the inverter/static transfer switch within 2 hours or be in at least SHUTDOWN within the next 24 hours, and

2. Reenergize the 120 VAC non-interruptible bus through the inverter/static transfer switch from its preferred source within 24 hours or be in at least SHUTDOWN within the next 24 hours.

c. With one 125 VDC bus not energized in the required manner from its associated battery and 1 dedicated battery charger, reenergize the 125 VDC bus from its associated battery and 1 dedicated battery charger within 24 hours* or be in at least SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified buses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the buses.

* A battery/battery charger can be disconnected from the distribution system for up to 5 consecutive days and for up to 10 cumulative days during a 92 day period, if an equalizing charge is being performed.

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AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.3 ON-SITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical buses shall be energized in the specified manner with tie breakers open between redundant buses:

a. Two essential 480 VAC buses energized from the associated 4160 VAC buses,**

b. Two 120 VAC non-interruptible buses energized through their respective inverter/static transfer switch from their preferred power sources, and

c. One 125 VDC bus energized from its associated battery and 1 dedicated battery charger.

APPLICABILITY: STARTUP*, SHUTDOWN*, and REFUELING

ACTION: With any of the above minimum required electrical buses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, control rod movements resulting in positive reactivity changes, or movement of IRRADIATED FUEL and initiate corrective action to reenergize the required buses as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified buses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the buses.

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* Whenever CALCULATED BULK CORE TEMPERATURE is less than or equal to 760 degrees F.

** The requirement for the tie breakers open between redundant buses is not applicable with respect to the 4160 VAC buses.

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BASIS FOR SPECIFICATION LCO 3.8.3/SR 4.8.3

The OPERABILITY and surveillance requirements of the on-site 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. The BASIS for Specifications 3.8.1 and 3.8.2 contains additional information.

Specification 3.0.5 provides the methodology and necessary data to determine the appropriate time interval to reach a CALCULATED BULK CORE TEMPERATURE of 760 degrees F. If the active core remains below this temperature, which corresponds to the design maximum core inlet temperature, then the design core inlet temperature cannot be exceeded and there can be no damage to fuel or PCRV internal components regardless of the amount, including total absence, or reversal, of PRIMARY COOLANT FLOW.

In ACTION 3.8.3.1.b.2, battery/battery charger 1C and in ACTION 3.8.3.1.c, battery/battery charger 1A or 1B can be inoperable for 5 consecutive days and 10 cumulative days during a 92 day period, if an equalizing charge is being performed. An equalizing charge per the manufacturer requires a nominal 3 to 5 days to complete and is required every 2 to 3 months. At FSV a battery/battery charger is disconnected from the distribution system when an equalizing charge is being performed to preclude damage to the DC loads. Due to the fact that a battery/battery charger can readily be placed back in service by reconnecting it to the system, this configuration is not considered a significant degradation of the DC power sources or the on-site distribution system.

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AUXILIARY ELECTRIC POWER SYSTEMS

3/4.8.4 ACM DIESEL GENERATOR

LIMITING CONDITION FOR OPERATION

3.8.4 The ACM Diesel Generator shall be OPERABLE with:

a. A flow path from any fuel oil storage tank through an OPERABLE fuel oil transfer pump to the ACM diesel fuel oil day tank,

b. A minimum of 10,000 gallons of fuel oil total in fuel oil storage tanks 1A and 1B (T-8401 and T-8402), and

c. The associated switchgear and motor control center OPERABLE.

APPLICABILITY: POWER, LOW POWER, STARTUP, and SHUTDOWN

ACTION:

a. With the ACM Diesel Generator inoperable, restore it to OPERABLE status within 7 days (not to exceed a total of 21 days in a three month period during STARTUP, LOW POWER, and POWER for performance of maintenance) or be in at least SHUTDOWN within the next 24 hours.

b. If results from the fuel oil sample taken per Surveillance Requirement 4.8.4.d are unacceptable, demonstrate the OPERABILITY of the ACM diesal generator by performing Surveillance Requirement 4.8.4.a.3 and Surveillance Requirement 4.8.4.a.4 within 8 hours, if not performed within the last 7 days. Resample the diesel fuel oil in fuel oil storage tanks 1A and 1B within 8 hours of receipt of the unacceptable sample results. If the results from the second sample are unacceptable, the ACM Diesel Generator shall be declared inoperable. Drain the fuel oil from the contaminated tank(s) and replace it with fresh fuel oil. Sample the fresh fuel oil per Surveillance Requirement 4.8.1.1.2.d.

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SURVEILLANCE REQUIREMENTS

4.8.4 The ACM diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.4-1 by:

1. Verifying that the water jacket heater is OPERABLE by ensuring the coolant water is being maintained at a temperature of greater than or equal to 85 degrees F,

2. Verifying the fuel oil transfer pump starts and transfers fuel from the storage system to the ACM diesel fuel oil day tank,

3. Verifying the ACM Diesel Generator starts, idles, and accelerates to the engine normal operating speed; the voltage shall be 4160 plus or minus 416 volts, and the frequency shall be 60 plus or minus 1.2 Hz,

4. Verifying the ACM Diesel Generator is synchronized, loaded to 2250 KW* plus or minus 250 KW, and operates at load for at least 60 minutes, and

5. Verifying the required fuel oil quantity in the fuel oil storage tanks.

b. At least once per 7 days by:

1. Verifying that the electrolyte level of each starting battery is above the plates, and

2. Verifying that the total battery terminal voltage is greater than or equal to 126 volts on float charge.

c. At least once per 31 days, and after each operation of the ACM diesel engines where the period of operation was greater than or equal to 1 hour, by checking for and removing accumulated water from the ACM Diesel Generator day tank.

d. At least once per 31 days, during POWER, by sampling the diesel fuel oil in fuel oil storage tanks 1A and 1B (T-8401, T-8402), in accordance with ASTM D2276-78 and verifying that the total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, method A.

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* This is a steady state load

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SPECIFICATION SR 4.8.4 (Continued)

e. At least once per 18 months by:

1. Subjecting the ACM Diesel Generator 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 4.8.4-2 can be energized via the ACM electrical distribution system.

3. Verifying that the ACM diesel fuel oil day tank level instrumentation functions properly to ensure fuel oil transfer pump operation.

4. Performing a CHANNEL CALIBRATION of the ACM Diesel Generator engine protective functions.

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TABLE 4.8.4-1

ACM DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests	Test Frequency
LTE 1	At least once per 31 days
GTE 2	At least once per 7 days*

* This test frequency shall be maintained until 7 consecutive failure-free demands have been performed and the number of failures in the last 20 demands has been reduced to 1 or less.

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TABLE 4.8.4-2

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 Pump	(P-4803 or P-4804)
i. Helium Purification Cooling Water Pump	(P-4701 or P-4702)
j. Firewater Pump House Vent Fan & Louvers	(C-7521 or C-7522)
k. Motor Operated Valve	(HV-2301 or HV-2302)
l. Stack Effluent Radiation Monitor	(RT-4801, RT-4802 and RT-4803)
m. ACM Plant Lighting	
n. Breathing Air Compressor	(C-4501 or C-4502)
o. ACM Battery Charger	(N-4888)

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BASIS FOR SPECIFICATION LCO 3.8.4/SR 4.8.4

The OPERABILITY of the ACM (Alternate Cooling Method) Diesel Generator ensures that a reliable and independent source of power will be available to equipment and instrumentation in the event of disruptive faults or events, such as an incapacitating fire in the THREE-ROOM CONTROL COMPLEX or other congested cable areas. The ACM Diesel Generator provides power independent of the normal plant electrical distribution system by manually repositioning electrical transfer switches. The ACM ensures that conditions analyzed and presented in FSAR Appendix D.1 for a permanent loss of forced circulation, are not exceeded in the case of such disruptive faults or events in congested cable areas. The ACM Diesel Generator is not SAFE SHUTDOWN COOLING equipment.

The specification is not applicable in REFUELING because the reactor is in a stable condition with low decay heat requirements, and therefore, low demands on the ACM supplied equipment.

A 7-day restoration time is required to permit maintenance and repair considering the limited availability of parts for the unique components of the ACM Diesel Generator. This time period is acceptable because of the limited nature of the specific events during which the ACM Diesel Generator is used.

The surveillance requirements are adequate for demonstrating the OPERABILITY of the ACM Diesel Generator to perform its intended function as described in FSAR Section 8.2.8.2. The testing of the ACM Diesel Generator simulates, where practical, the parameters of operation that would be expected if an actual demand was to be placed on the system. Weekly testing of the startup batteries ensures that adequate voltage is available to start the ACM Diesel Generator upon demand. The check that the water jacket heater is OPERABLE, increases the probability of a successful start during cold weather conditions. The check that the fuel oil system is capable of transferring fuel oil to the ACM diesel fuel oil day tank ensures fuel oil is available for ACM Diesel Generator operation. The testing of the ACM Diesel Generator demonstrates proper startup and load-carrying capability, and verifies that the required voltage and frequency are attained. This test also verifies that the components of the ACM Diesel Generator are OPERABLE.

The "ACM Diesel Generator" 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 ACM Diesel Generator to the ACM components upon demand.

BASIS FOR SPECIFICATION LCO 3.8.4/SR 4.8.4 (Continued)

At least 10,000 gallons of fuel oil is required to be available in fuel oil storage tanks 1A and 1B. This requirement is normally met by satisfaction of Specification 3.8.1.1.b.2. During STARTUP and SHUTDOWN with a CALCULATED BULK CORE TEMPERATURE less than 760 degrees F, the 10,000 gallon requirement of this specification assures OPERABILITY of the ACM diesel generator. Ensuring that a fuel oil transfer system is OPERABLE with 10,000 gallons of fuel provides for over 72 hours of ACM Diesel Generator operation with a full ACM load of 900 KW. This is considered more than adequate time for obtaining additional fuel from off-site sources.

The requirements on the fuel oil system provide adequate assurance that the ACM Diesel Generator will continue to supply reliable electric power for the duration of the ACM event. With over 10 years experience, there have been no fuel oil related failures of the ACM Diesel Generator. Fuel oil is distributed to the ACM Diesel Generator between a shared tank arrangement with the Auxiliary Boilers. Diesel Fuel Oil Systems tank and related piping arrangements are shown in Figure 3.8.1-1. The turnover of the fuel oil in the Auxiliary Boiler fuel oil storage tanks during SHUTDOWN, STARTUP, and LOW POWER and the performance of Surveillance Requirement 4.8.4.d during POWER ensure the quality of fuel oil in the underground storage tanks and the ACM fuel oil day tank.

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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 *, and

c. The SHUTDOWN MARGIN requirements of Specification 3.1.3 shall be met.

APPLICABILITY: Whenever both primary and secondary PCRV closures of any PCRV penetration are removed

ACTION:

a. With the conditions of a or b above not met, restore the condition(s) to within the above limits within 1 hour, or terminate fuel handling and vessel internal maintenance retract the fuel handling mechanism or any other remotely operated mechanisms from the PCRV, and close the reactor isolation valve or opening through the PCRV as soon as practicable.

b. With the SHUTDOWN MARGIN requirements of Specification 3.1.3 not met, comply with ACTION c of Specification 3.1.3.

SURVEILLANCE REQUIREMENTS

4.9.1 a. The reactor pressure and temperature conditions shall be determined to be within the above limits at least once per 12 hours,

b. Verification of SHUTDOWN MARGIN shall be in accordance with Specification 4.1.3.

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* Applicable only when the fuel handling machine is located on the reactor vessel, with the cask isolation valve and reactor isolation valve open.

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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 short-term 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.

The ACTION statement ensures that reactor and fuel handling machine will be placed in the safest configuration as soon as practicable, if a required condition cannot be maintained.

The surveillance requirement frequency gives adequate assurance that changes in reactor conditions will be detected in time to permit corrective actions if required.

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 The reactivity of the core shall be continuously monitored by at least two startup channel neutron flux monitors.

APPLICABILITY: REFUELING

ACTION:

a. With one of the above required neutron flux monitors inoperable, or not operating, immediately suspend all operations involving CORE ALTERATIONS, any evolution resulting in positive reactivity changes, or movement of IRRADIATED FUEL.

b. With both of the above required neutron flux monitors inoperable or not operating:

1. Immediately suspend all operations involving CORE ALTERATIONS, any evolution resulting in positive reactivity changes, or movement of IRRADIATED FUEL,

2. Retract the fuel handling mechanism or any other remotely operated mechanism from the PCRV,

3. Close the reactor isolation valve or opening through the PCRV as soon as practicable, and

4. Within 12 hours evaluate the SHUTDOWN MARGIN per Specification 4.1.3.

SURVEILLANCE REQUIREMENTS

4.9.2 Each startup channel neutron flux monitor shall be demonstrated OPERABLE:

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

b. Within 24 hours prior to the initial start of CORE ALTERATIONS, by performance of a CHANNEL FUNCTIONAL TEST, and

c. At least once per 7 days, by performance of a CHANNEL FUNCTIONAL TEST.

BASIS FOR SPECIFICATION LCO 3.9.2 / SR 4.9.2

The OPERABILITY of the startup channel neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. (Additional information is contained in the BASIS for Specification 3.3.1). In accordance with the definition of the REFUELING MODE, this specification also applies to reactor internal maintenance that does not involve a CORE ALTERATION, such as a helium circulator changeout or modifications or repair of the primary coolant purification systems. For this maintenance where no changes are being made to core reactivity, certain activities are permitted in the event of inoperable startup channel flux monitors, provided the other ACTION requirements are met.

The ACTION statement ensures that activities that could affect the reactivity condition of the core are suspended whenever neutron flux monitoring capabilities are degraded.

The surveillance requirements ensure that the neutron flux monitors are capable of detecting changes in reactor conditions in time to permit corrective actions if required. These are i- addition to the calibration requirements of Specification 4.3.1.

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.3 FUEL HANDLING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.3 The Fuel Handling Machine (FHM) shall be OPERABLE with:

a. A helium* pressure in the FHM at atmospheric pressure or slightly below,

b. The fuel handling purge system connected and OPERABLE, with a supply of helium* available, and the gas waste system available to receive purge gas,

c. One cooling water coil 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 FHM is bolted to a reactor isolation valve over the reactor, a fuel storage well, or the fuel shipping cask loading port, and

e. Switches and alarms which verify correct orientation and placement of fuel elements OPERABLE.

APPLICABILITY: During use of the FHM for reactor internal maintenance or any handling of IRRADIATED FUEL*.

---

* A helium atmosphere is not required in the FHM when it is empty or when handling unirradiated fuel or handling IRRADIATED FUEL that has undergone fission product decay for 100 days or more, or when the FHM is not being used in fuel handling operations with the reactor core or fuel storage facility.

SPECIFICATION LCO 3.9.3 (Continued)

ACTION:

a. With requirements a, b, or d above not satisfied, within 1 hour initiate action to retract the fuel transfer mechanism and close the reactor isolation valve and FHM cask valve.

b. With requirement a above not satisfied when the FHM cask valve is closed, within 1 hour initiate action to reduce gas inflow to the FHM and/or increase purge gas outflow to the gas waste system, as appropriate.

c. With requirement c above not satisfied, within 1 hour initiate action to return the IRRADIATED FUEL elements within the FHM to the reactor core or to the fuel storage facility, and terminate fuel handling with the FHM.

d. With requirement e above not satisfied, terminate placement of the fuel elements within the reactor vessel with the FHM.

SURVEILLANCE REQUIREMENTS

4.9.3.1 At least once per 12 hours, the FHM shall be demonstrated OPERABLE by verifying that:

a. The internal pressure is within the above limit, and

b. The cooling coil water outlet temperature is within the above limit.

4.9.3.2 Within 31 days prior to beginning refueling operations, the FHM shall be demonstrated OPERABLE by:

a. Functionally testing the FHM, FHM cask valve and reactor isolation valves, and by performing a CHANNEL FUNCTIONAL TEST of the interlocks, limit switches, and alarms,

SPECIFICATION SR 4.9.3.2 (Continued)

b. Functionally testing the redundant cooling water coils and verifying:

1. The flow rates are sufficient to silence the low-flow alarms, and

2. The outlet temperatures are less than 150 degrees F.

c. Performing a CHANNEL CALIBRATION of the cooling water temperature alarms,

d. Functionally testing the FHM cooling water leak detector by adding water to the drain header,

e. Visually inspecting the backup fire water connections and hose to verify the hose and connector are in place and undamaged, and

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 by evacuating and filling the FHM.

4.9.3.3 Within 24 hours prior to initial attachment to the FHM, and at least once per 14 days thereafter, the reactor building overhead crane shall be demonstrated OPERABLE by:

a. Performing a visual inspection of crane components, including cable and reeving, blocks and sheaves, and the special lifting hook for carrying the FHM when it is installed and secured on the crane, and

b. Functionally testing the upper limit switch of the FHM hoist under no load by slowly running the block into the limit switch.

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* A helium atmosphere is not required in the FHM when it is empty or when handling unirradiated fuel or handling IRRADIATED FUEL that had undergone fission product decay for 100 days or more, or when the FHM is being used in fuel handling operations with the reactor core or fuel storage facility.

BASIS FOR SPECIFICATION LCO 3.9.3/SR 4.9.3

The objective of this specification is to ensure the prevention of any uncontrolled release of radioactivity during handling of IRRADIATED FUEL or reactor internal maintenance, and to ensure the correct installation of fuel elements into the reactor core and the fuel storage wells.

The OPERABILITY requirements for the FHM ensure that:

a. 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,

b. The FHM is connected to the gas waste system, and can be purged with helium while 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,

c. Fuel elements contained in the FHM 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 (FSAR Section 14.6.3.1). Silencing of the low-flow alarm ensures existence of cooling water flow, while the 150 degrees F limit ensures adequacy of flow, and

d. A seismic event will not adversely affect the FHM when it is bolted to a reactor isolation valve or supported by the reactor building overhead crane.

The ACTION statements ensure that the FHM will be promptly placed in a safe configuration in the event any of the OPERABILITY requirements can no longer be met during REFUELING. For example, if a switch or alarm necessary to verify correct seating of fuel elements becomes inoperable, the ACTION statement prohibits continuing seating of the fuel elements in the core. However, it does not prohibit unloading the FHM into a fuel storage well where exact orientation is not a safety concern. This would permit the FHM to be unloaded for easier repair and minimize radiation exposure to maintenance personnel.

BASIS FOR SPECIFICATION LCO 3.9.3/SR 4.9.3 (Continued)

The FHM provides for the safe refueling of the reactor. To ensure the reliability of the FHM during the refueling operation, the FHM 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 fire water connections and hose will be made. These checks ensure the capability to maintain the proper atmosphere environment within the FHM, to prevent any uncontrollable release of activity, and the capability to maintain the surface temperature of fuel elements within the FHM below 750 degrees F.

The reactor building overhead crane is inaccessible for visual inspections and tests during operation with the FHM. It must be moved to an accessible location to perform the surveillances identified. Many of the crane operations involve several days when it is attached to the FHM and is not moved. Inspections once per 14 days are appropriate for assuring safe operation of the reactor building overhead crane.

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.4 FUEL STORAGE WELLS

LIMITING CONDITION FOR OPERATION

3.9.4 Each fuel storage well containing IRRADIATED FUEL shall have a helium atmosphere 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 coil operating 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 9000 cfm through the fuel storage facility.

c. IRRADIATED FUEL shall not be located in the central column of a fuel storage well.

APPLICABILITY: When IRRADIATED FUEL is stored 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 perform one of the following within the next 24 hours:

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, or

c. Perform an engineering evaluation to confirm that the hottest fuel element surface temperature will not exceed 750 degrees F.

SURVEILLANCE REQUIREMENTS

4.9.4 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 that the outlet cooling water temperature of operating cooling coil(s) is 150 degrees F or less, with a flow rate of greater than 7 gpm, and either:

a) Both cooling water coils are OPERABLE with at least one coil in operation, or

b) One cooling water coil is OPERABLE and in operation, and the fuel storage well emergency booster fan is OPERABLE.

2. Verifying that the pressure within the well is at atmospheric pressure or slightly below.

b. At least once per 31 days by verifying that the fuel storage well emergency booster fan operates upon manual initiation.

c. At least once per REFUELING CYCLE, by verifying the capability of the fuel storage well emergency booster fan to draw a minimum of 9000 cfm of air through the fuel storage facility.

d. Verifying during fuel handling that no IRRADIATED FUEL elements are inserted into the central position of fuel storage wells.

BASIS FOR SPECIFICATION LCO 3.9.4/ SR 4.9.4

The storage well cooling water system is designed with two 100% capability cooling coils supplied from independent water sources (FSAR Section 9.1 2).

The accident conditions described in the FSAR postulate the total loss of cooling water 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. An outlet temperature of 150 degrees or less from a single cooling coil ensures the hottest IRRADIATED FUEL element surface temperatures 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.

The ACTION statements provide for corrective actions within an adequate time to prevent the hottest stored fuel element surface temperature from reaching 750 degrees F, thus obviating any significant oxidation in the unlikely combined events of air inleakage and loss of normal cooling.

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.5 SPENT FUEL SHIPPING CASK

LIMITING CONDITION FOR OPERATION

3.9.5 A spent fuel shipping cask shall not be loaded or partially loaded with IRRADIATED FUEL having less than 100 days of fission product decay.

APPLICABILITY: At all times

ACTION:

a. IRRADIATED FUEL with less than 100 days of fission product decay shall be removed from the spent fuel shipping cask prior to lifting the cask.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 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.

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BASIS FOR SPECIFICATION LCO 3.9.5/SR 4.9.5

The potential radiological consequences of an accident whereby the spent fuel shipping cask breaks open while being lowered to the truck, have been analyzed in FSAR Section 14.6.3.3. This analysis assumes that the cask is loaded with the most radioactive IRRADIATED FUEL elements contemplated to be shipped from the plant after 100 days of fission product decay. Determination prior to loading the cask that the IRRADIATED FUEL elements have undergone at least 100 days of fission product decay ensures that a cask loading accident is within the assumptions used in the safety analysis.

FUEL HANDLING AND STORAGE SYSTEMS

3/4.9.6 COMMUNICATIONS DURING CORE ALTERATIONS

LIMITING CONDITION FOR OPERATION

3.9.6 Direct two-way communications shall be maintained between operations control room personnel and personnel at the Fuel Handling Machine (FHM) control room.

APPLICABILITY: During CORE ALTERATIONS*

ACTION: With no direct communications between the operations control room personnel and personnel at the FHM control room, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.6 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*.

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* Except for withdrawal of control rod pairs for testing, when no other CORE ALTERATION activities are in progress.

BASIS FOR SPECIFICATION LCO 3.9.6/4.9.6

The requirement for communications 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 so that core conditions can be monitored.

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. Maintaining direct communication also permits the operations control room to immediately request a stop of any movements causing excessive count rate changes.

The surveillance times specified give adequate assurance that communications will be available as needed.

During withdrawal of control rod pairs for testing, which is by definition a CORE ALTERATION, sufficient controls and indications are present for operations control room personnel so that communications with FHM control room personnel are not required.

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SPECIAL TEST EXCEPTIONS

3/4.10.1 XENON STABILITY

LIMITING CONDITION FOR OPERATION

3.10.1 The requirements of Specification 3.1.4.1.a and b may be suspended during the performance of the Xenon Stability PHYSICS TESTS, provided power perturbations do not exceed those which are expected, as determined by an engineering evaluation performed prior to the test and approved by the NFSC.

APPLICABILITY: POWER*

ACTION: With the occurrence of power perturbations in excess of the approved maximum limits, reduce THERMAL POWER sufficiently to stop power perturbations.

SURVEILLANCE REQUIREMENTS

4.10.1 There is no specific surveillance requirement associated with the above LCO. This test is conducted in accordance with the appropriate B series startup test procedures.

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* During Xenon Stability PHYSICS TESTS

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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 attempt to initiate xenon oscillations by inserting an out-of-sequence rod. The limits on the average temperature rise in LCO 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.

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 3.1.4.1.a and b throughout the test. This engineering evaluation will assure that the maximum worth rod pair criteria are met (FSAR Sections 14.2.2.6 and 3.5.3.1), and that the assumptions utilized in the development of Core Safety Limit SL 2.1.1 (with regard to region, column, and axial peaking factors) remain valid throughout the test.

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SECTION 5.0

DESIGN FEATURES

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DESIGN FEATURES

5.1 ___

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 (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 for the exclusion area. There are no permanent residences located within the exclusion area.

An information center is located within the exclusion area, but outside of 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.

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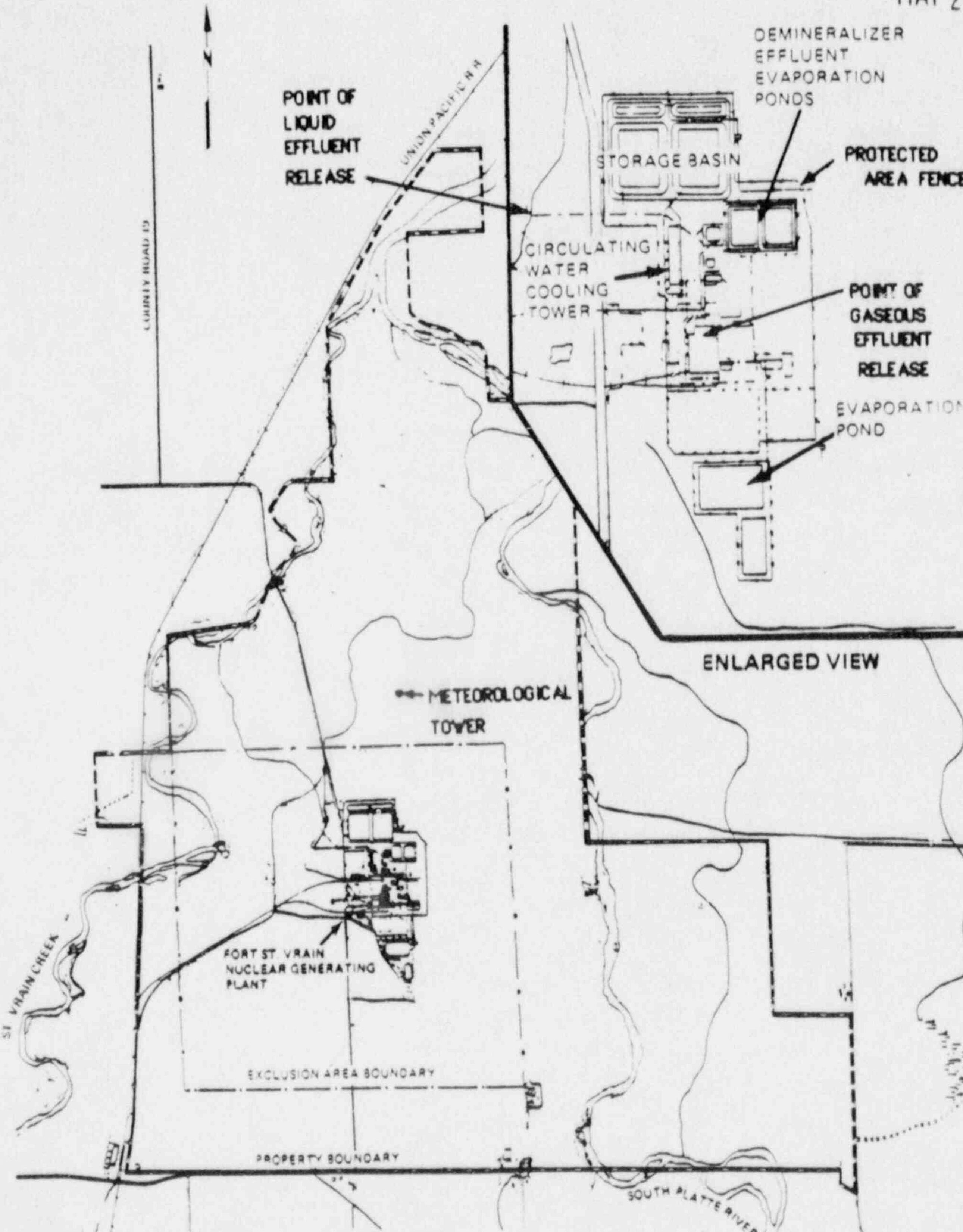

SITE OF THE FORT ST. VRAIN
NUCLEAR GENERATING STATION

Figure 5.1-1

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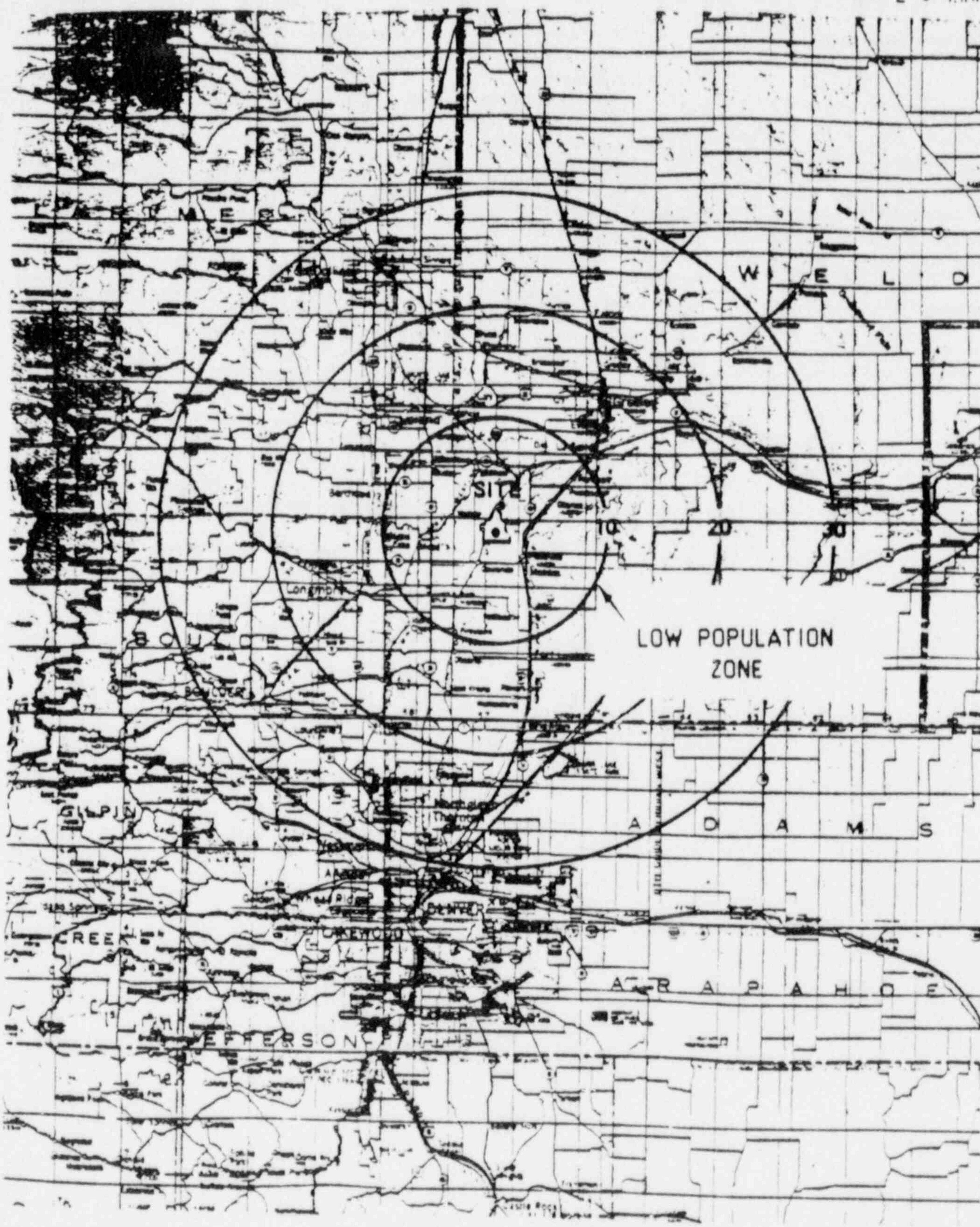

AREA WITHIN THIRTY MILES OF SITE
INCLUDING LOW POPULATION ZONE

Figure 5.1-2

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.

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 liner, contain the radioactive primary coolant in a manner analogous to a conventional primary vessel. The secondary closures enclose the primary closures and together with the PCRV structure, contain the radioactive primary coolant that might be released from a leaking primary boundary 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 (FSAR Section 6.6) 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 Sections 6.5 and 14.5.

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 was 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 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, up to eight test fuel elements have resided in the reactor core, depending on which fuel segments are included in a given fuel cycle. 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.

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 distributed 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).

The new fuel and burnable poison loading for each reload segment shall satisfy the following requirements:

a. Provide adequate core reactivity for burnup during each cycle,

b. Ensure an acceptable SHUTDOWN MARGIN throughout the cycle with any one control rod pair withdrawn with the core at a average temperature of 80 degrees F, and

c. Ensure an acceptable SHUTDOWN MARGIN throughout the cycle with any two control rod pairs withdrawn with the core at an average temperature of 220 degrees F for at least two weeks.

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To satisfy the criteria for reactor power distribution and maximum control rod worth, each REFUELING CYCLE has a control rod withdrawal sequence that is specified for use during operation.

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 those described for the worst case rod-pair withdrawal accident in the AEC Safety Evaluation of Fort St. Vrain dated January 20, 1972.

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 degrees 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

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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*

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	All regions	Less than or equal to 1.61*

* An uncertainty of plus or minus 10% for the Axial Peaking Factor and plus or minus 15% for the Intra-Region Peaking has been assumed in the fuel failure analysis when using these calculated values. This assumes that this calculated value is within the limits stated herein, and that the actual power peaking is within the assumed uncertainties.

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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 elements above the hexagonal columns contains 1 weight percent crushed boronated graphite. The top layer of reflector elements above the permanent side reflector blocks contains 1 weight percent boronated graphite enclosed in steel cans.

5.3.6 Control Rods

There are 37 control rod pairs, each made up of annular cylindrical absorber bodies containing boron carbide in a graphite matrix sheathed in metal cans. The nominal boron load is 0.48 grams/cc of compact in the inner 19 control rod pairs and 0.63 gram/cc of compact in the outer 18 control rod pairs. These boron loads correspond to boron weight fractions of about 0.30 and 0.40, respectively. The absorber compacts with a lower boron concentration have a greater high temperature stability than the heavier loaded compacts, which is of benefit during a complete loss of forced cooling accident.

The two members of a rod pair are separated by 9.72 inches and operate within 4 inch O.D. holes provided in the central fuel element column in each refueling region. Each control rod pair has an independent drive mechanism. The rods will operate in a bank only during a scram or runback. All control rod pairs have scram capability and are driven by gravity. The total insertion time is approximately 152 seconds. Complete withdrawal takes approximately 180 seconds.

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With the exception of the central rod pair, which is used for automatic regulation, the control rods during operation are usually inserted only about 50% of the time or less. The distribution of fuel of different ages requires the use of alternate shim rods each year of full power operation. The depletion of a significant fraction of the boron has a relatively small influence on the rod worth. For example, if a rod were inserted constantly for 6 full power years, approximately 20% of the Boron-10 would be lost by neutron capture, but the reactivity worth of the rod would decrease by less than 5% of its original value. Therefore, from a neutronic standpoint, it would not be necessary to replace each control rod as frequently as every 6 full power years. However, due to other irradiation effects each control rod pair is replaced after six years of full power operation.

5.3.7 Reserve Shutdown System

The reserve shutdown system design basis is described in the BASIS for Specification 3/4.1.6. The reserve shutdown system consists of boronated graphite balls, about 0.5 inches in diameter, which can be introduced into the core via cylindrical holes in the central fuel column of each refueling region. The balls for the central 19 refueling regions contain 0.32 gram/cc of boron in each ball (approximately 20% boron carbide-B4C by weight) while the balls for the outer 18 refueling regions contain 0.63 gram/cc of boron (approximately 40% boron carbide by weight). As with control rod boron loadings, the reserve shutdown system has lighter boron loads in the central refueling region to enhance the stability of the poison material in the hottest core regions during a permanent Loss Of Forced Circulation accident.

The nominal reactivity worth of the reserve shutdown system is at least 0.14 delta k in the initial core and at least 0.13 delta k during the equilibrium core, in absence of any control rods.

The reserve shutdown system by itself is capable of safely shutting down the reactor to refueling temperature and maintaining it subcritical for an extended period of time. In addition, the reserve shutdown system can be used to augment the shutdown margin by the insertion of the reserve shutdown balls in those core regions where control rods failed to insert.

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 Typ	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 el ment CIB - HEU
UC(x)O(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/2	TRISO & ThO/2 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.

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* C/2 and O/2 mean dicarbide and dioxide, respectively.

** x and y represent the mean quantities of carbon and oxygen and do not signify a specific compound. These values are explicit in the final fuel specification. All kernels of this type are derived from WAR beads.

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5.4 FUEL STORAGE

5.4.1 Criticality

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.

5.4.2 Containment

The fuel storage wells are designed and shall be maintained to have gas- and water-tight containment.

5.4.3 Neutron Absorber-Heat Sink

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 (iron slag) to act as a built-in neutron absorber and heat sink.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower is 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 maintained within the cyclic limits of Table 5.6-1 when evaluated against the identified plant transients of Table 5.6-1.

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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	470					
External Main Steam Pipe		700				
Internal Main Steam Bypass Piping			1000			
- 8" X 16" latrolet				123		
- bypass line			550		175	
Hot Reheat Steam Piping			560			185

(1) Transient Identification per Design Criteria DC-5-2, Transient Conditions:

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

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.0 ADMINISTRATIVE CONTROLS

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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 control room and Shift Supervisor's office, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President, Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

Offsite and Onsite Organizations

6.2.1 Onsite and Offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.

b. The Station Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President, Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

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d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

Unit Staff

6.2.2

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, reactor 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 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.

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 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

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* Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

f. 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 unforeseen 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:

1. An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

2. 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).

3. A break of at least 8 hours should be allowed between work periods (including shift turnover time).

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on shift.

g. If unusual circumstances arise requiring deviation from the above guidelines, such deviation shall be authorized by the Station Manager or his designees, 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 excessive individual overtime hours have not been assigned. Routine deviation from the above guidelines is not authorized.

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Authorized deviations to the working hour guidelines shall be documented and available for review by the Commission.

h. The Shift Supervisors and either the Station Manager or the Superintendent of Operations shall hold a Senior Reactor Operator's license. The Reactor Operators shall hold a Reactor Operator's license.

In addition to the requirements of Section 6.2.2, the following apply:

During SHUTDOWN or REFUELING, an individual (other than the Technical Advisor) with a valid RO (or SRO) license shall be present in t e 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.

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 2 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 or Shift Supervisor's office while the unit is in POWER, LOW POWER or STARTUP, an individual (other than the Technical Advisor) with a valid SRO license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room or Shift Supervisor's office while the unit is in SHUTDOWN or REFUELING, an individual with a valid SRO license or RO license shall be designated to assume the control room command function.

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TABLE 6.2-1
MINIMUM SHIFT CREW COMPOSITION

Position	Number of Individuals Required to Fill Position: During SHUTDOWN or REFUELING	Number of Individuals Required to Fill Position: All Other Conditions
SS (SRO)	1	1
SRO	Not Required	1
RO	1	2 (a)
EO	1	1
AT	Not Required	1
TA (b)	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)

NOTES TO ACCOMPANY TABLE 6.2-1

a. 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.

b. The Technical Advisor shall be scheduled as required by Specification 6.2.3.2.e.

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Technical Advisors

6.2.3 The function, responsibilities and authority of the Technical Advisors shall be as follows:

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.

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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 (ANSI) 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.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR 55. Compliance with Section 5.5 of ANSI 18.1-1971 shall be achieved no later than 6 months following commencement of commercial operation.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Nuclear Training Manager 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's 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 Nuclear Training Manager. The Technical Advisors shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room.

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6.5 REVIEW AND AUDIT

6.5.1 Plant Operations Review Committee (PORC).

Function

6.5.1.1 The PORC shall function to advise the Manager of Nuclear Production on all matters related to nuclear safety.

Composition

6.5.1.2 The PORC shall be composed of the following:

Chairman: Station Manager
Technical/Administrative Services Manager
Support Services Manager (Radiation Protection Manager)
Superintendent of Operations
Superintendent of Maintenance
Superintendent of Nuclear Betterment Engineering
Results Engineering Supervisor
Administrative Shift Supervisor
Superintendent of Technical Services Engineering
Nuclear Training Manager

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 as voting members 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 or his designated alternate.

Quorum

6.5.1.5 A quorum shall consist of the Chairman or alternate Chairman, and four members including alternates.

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Responsibilities

6.5.1.6 The PORC shall be responsible for:

a. Review of all procedures required by Technical Specification 6.8.1, 6.8.2, 6.8.3, and 6.8.4 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.

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. Review of investigations 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 NFSC.

f. Review of all REPORTABLE EVENTS.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses, and reports thereon as requested by the Chairman of the NFSC.

i. Review of the facility Security Plan and implementing procedures.

j. Review of the facility Radiological Emergency Response Plan and implementing procedures.

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k. Review of changes to the Process Control Program, the Off-site Dose Calculation Manual, and the Radwaste Treatment System.

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 prior to their implementation, with the exception of Procedure Deviation Reports.

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 written notification within 24 hours to the Vice President, Nuclear Operations 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 6.1.1 above.

Records

6.5.1.8 The PORC shall maintain written minutes of each meeting that, at a minimum, document the results of all PORC activities performed under the responsibility provisions of the Technical Specifications. Copies shall be provided to the Manager, Nuclear Production and Chairman of the Fort St. Vrain NFSC.

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6.5 2 Nuclear Facility Safety Committee (NFSC)

Function

6.5.2.1 The NFSC shall collectively have the competence required to review problems in the following areas:

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: As appointed, see 6.5.2.4
Manager, Nuclear Licensing and Fuels Division
Manager, Nuclear Production Division
Manager, Nuclear Engineering Division
Manager, Quality Assurance Division
Safety and Security Director
Consultants, as required and appointed
by the Vice President, Nuclear Operations

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Alternates

6.5.2.3 Alternate members, if required, shall be appointed in writing by the Vice President, Nuclear Operations; however, no more than two alternate members shall participate as voting members in NFSC activities at any one time.

Chairman

6.5.2.4 The Chairman and Alternate Chairman of the NFSC shall be appointed in writing by the Vice President, Nuclear Operations and shall serve as members of the NFSC.

Consultants

6.5.2.5 Consultants, as required, shall be appointed in writing by the Vice President, Nuclear Operations, to provide expert advice to the NFSC.

Meeting Frequency

6.5.2.6 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 6 months thereafter.

Quorum

6.5.2.7 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.

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Review

6.5.2.8 The NFSC 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 10 CFR 50.59, 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 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question, as defined in 10 CFR 50.59.

d. Proposed changes in Technical Specifications or this Operating License.

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.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated 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.

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6.5.2.9 The NFSC shall approve:

a. The control rod withdrawal sequence as required by Technical Specification LCO 3.1.4 and DF 5.3.4.

b. The calculated base reactivity curve for use with each cycle as well as any changes to this data, including the justification of the change during the cycle, as discussed in the BASIS for LCO 3.1.7. NFSC permission is required before reactor operations may resume if a reactivity anomaly of 0.01 delta k is reached.

c. Proposed changes to facility, operating procedures and tests or experiments that are determined to involve an unreviewed environmental question.

d. Engineering evaluations that identify power pertubations expected during Xenon Stability PHYSICS TESTS. See Specification 3.10.1.

Audits

6.5.2.10 Audits of facility activities shall be performed under the cognizance of the NFSC. These audits shall encompass:

a. The conformance of facility operation to 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 6 months.

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

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e. The review of the facility Radiological Emergency Response Plan and implementing procedures at least every 12 months by persons who have no direct responsibility for implementation of the 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 Plan.

g. Any other area of facility operation considered appropriate by the NFSC.

h. With respect to fire protection:

   1. an annual fire protection and loss prevention inspection and audit utilizing either qualified off-site licensee personnel or an outside fire protection firm,

   2. a biennial audit of the fire protection program and implementing procedures,

   3. a triennial fire protection and loss prevention inspection and audit utilizing an outside qualified fire consultant.

i. The Off-site 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 for effluent and environmental monitoring, at least once per 12 months.

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Authority

6.5.2.11 The NFSC shall report to and advise the Vice President, Nuclear Operations in those areas of responsibility specified in 6.5.2.8, 6.5.2.9 and 6.5.2.10.

Records

6.5.2.12 Records of NFSC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NFSC meeting shall be prepared and forwarded to the Vice President, Nuclear Operations within 30 days following each meeting.

b. After preliminary approval by the Vice President, Nuclear Operations, the minutes shall be distributed to all NFSC members and approved at the next NFSC meeting.

c. Reports of reviews encompassed by Section 6.5.2.8 shall be forwarded to the Vice President, Nuclear Operations within 30 days following completion of the review.

d. Audit reports encompassed by Section 6.5.2.10 shall be prepared and forwarded to the Vice President, Nuclear Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

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6.6 REPORTABLE EVENTS ACTION

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.72 and 10 CFR 50.73, and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and submitted to the NFSC and Station Manager.

6.7 SAFETY LIMIT VIOLATIONS

If a SAFETY LIMIT is exceeded, as defined in Specifications 2.1 and 2.2, the following actions shall be taken:

a. In accordance with 10 CFR 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 Chairman, NFSC shall be notified within 24 hours.

b. A Licensee Event Report shall be prepared in accordance with 10 CFR 50.73.

c. The License Event Report shall be submitted to the Commission in accordance with 10 CFR 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 Commission.

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 Safety Guide 33, November 1972.

b. Refueling operations.

c. Surveillance and test activities (except post-maintenance tests) of safety related equipment.

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d. Security plan implementation.

e. Radiological Emergency Response Plan implementation.

f. Fire protection program implementation.

g. Process Control Program (PCP) implementation.

h. Off-site Dose Calculation Manual (ODCM) implementation.

i. Quality Assurance Program for effluent and environmental monitoring.

j. Fuel surveillance program implementation.

k. Personnel radiation protection.

l. Inservice Inspection and Testing program implementation.

6.8.2 Procedures of Specification 6.8.1, and changes thereto, shall be reviewed by the PORC, or a subcommittee thereof, and approved by the appropriate management prior to implementation. Procedures shall be reviewed periodically as set forth in Administrative Procedures.

The post-maintenance test program for safety related equipment shall be reviewed by PORC.

Security Plan procedures, and changes thereto, shall be reviewed by the PORC, or a subcommittee thereof, and approved by the designated Plant Security Officer prior to implementation.

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6.? . Temporary changes to procedures of 6.8.1 above may be ?a? provided:

a e intent of the original procedure is not altered,

.. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License, and

c. The change is documented, reviewed by the PORC, or a subcommittee thereof, and approved by the appropriate management 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 conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. 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:

1. Identification of a sampling schedule for the critical variables and control points for these variables,

2. Identification of the procedures used to measure the values of the critical variables,

3. Identification of process sampling points, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4. Procedures for the recording and management of data,

5. Procedures defining corrective actions for all off-control point chemistry conditions, and

6. A procedure identifying (a) the 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 include the following:

1. Training of personnel,

2. Procedures for sampling and analysis, and

3. Provisions for maintenance of sampling and analysis equipment.

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6.9 REPORTING REQUIREMENTS

6.9.1 Routine Reports

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 Regional Office unless otherwise noted.

Startup Report

6.9.1.1 a. 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. The final Startup Report shall be submitted within 90 days following completion of the startup test program.

Annual Reports and Summary

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 Occupational 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 by 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.

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* This tabulation supplements the requirements of 10 CFR 20.407

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b. Annual Primary Coolant Activity Report

The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.2 shall be submitted by March 31 of each year. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioiodine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

c. 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) by May 1 of each year.

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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.

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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 were not achievable.

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 Commission in the Annual Radiological Environmental Monitoring Report.

d. Annual Diesel Generator Reliability Report

An annual data report on Standby Diesel Generator reliability shall be submitted to the Commission in a manner consistent with Regulatory Guide 1.108 position C.3.a prior to March 1 of each year.

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* One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

Semi-Annual Radioactive Effluent Release Report

6.9.1.3 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 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 Radioactive Effluent Release Report submitted by March 1 of each year shall include a summary assessment for the previous calendar year 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 5.1-1) 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 Off-site Dose Calculation Manual (ODCM).

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The Radioactive Effluent Release Report submitted by March 1 for the previous calendar 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 40 CFR 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.

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste as defined by 10CFR61 shipped off-site 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.

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The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and the OFF-SITE DOSE CALCULATION MANUAL, pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. They shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification ELCO 8.2.1.

The Radioactive Effluent Release Reports shall also include the following: An explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification ELCO 8.1.2 or ELCO 8.1.1, respectively; and description of the events leading to liquid hold-up tanks or gas storage tanks exceeding the limits of Specification ELCO 8.1.2 or ELCO 8.1.1, respectively.

Monthly Operating Report

6.9.1.4 A routine Monthly Operating Report covering the operation of the plant during the previous month shall be submitted prior to the fifteenth calendar day of the following month. Submittal shall be to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D. C. 20555, 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.

Special Reports

6.9.2 Special Reports shall be submitted to the NRC Regional Administrator within the time period specified for each report.

6.9.3 Non-Routine Radiological Reports

a. Radioactive Gaseous Effluent

1. If the calculated dose from the release of gaseous effluents pursuant to ESR 8.1.1.i exceeds any of the limits in ELCO 8.1.1.h, in lieu of a Licensee Event Report, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to ensure that subsequent releases will be in compliance with the above limits will be prepared and submitted to the Commission within 30 days.

2. If gaseous waste is discharged without treatment and in excess of the limits, in lieu of a Licensee Event Report, a Special Report that includes the following information shall be prepared and submitted to the Commission within 30 days:

a) Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

b) Action(s) taken to restore the inoperable equipment to operable status, and

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

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b. Radioactive Liquid Effluent

1. If the calculated dose from the release of radioactive materials in liquid effluents pursuant to ESR 8.1.2.e exceeds any of the limits specified in ELCO 8.1.2.g, in lieu of a Licensee Event Report, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits will be prepared and submitted to the Commission within 30 days.

2. If radioactive liquid waste is discharged without treatment pursuant to ELCO 8.1.2.h, and in excess of the limits, in lieu of a Licensee Event Report, a Special Report that includes the following information shall be prepared and submitted to the Commission within 30 days:

   a) Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   b) Action(s) taken to restore the inoperable equipment to OPERABLE status, and

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

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c. Radioactive Effluents - Total Dose

If the limits of ELCO 8.1.5.a have been exceeded, in lieu of a Licensee Event Report, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits shall be prepared and submitted to the Commission within 30 days. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190. Submittal of the Report is considered a timely request, and a variance is granted until staff action on the request is complete.

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d. Radiological Environmental Monitoring

If the level of radioactivity as a result of plant effluents in an environmental sample medium at a specified location exceeds the reporting levels of Table 8.2-3 of ELCO 8.2.1, when averaged over any calendar quarter, in lieu of a Licensee Event Report, pursuant to Specification ELCO 8.2.1.c, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents such that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications ELCO 8.1.1.h and ELCO 8.1.2.g will be prepared and submitted to the Commission within 30 days. When more than one of the radionuclides in Table 8.2-3 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{Reporting Level (1)}} + \frac{\text{Concentration (2)}}{\text{Reporting Level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 8.2-3 are detected and are the result of plant effluents, a Special Report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications ELCO 8.1.1.i and ELCO 8.1.2.g. This Special Report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Monitoring Report.

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.

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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,

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications,

e. Records of changes made to the procedures required by Specification 6.8.1,

f. Records of sealed source leak tests and results,

g. Records of annual physical inventory of all source material of record, and

h. Annual Meteorological Data Summary - An annual summary of hourly meteorological data collected over the previous year shall be maintained for 5 years. This annual summary may be either in the form of an hour-by hour listing or 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.

6.10.2 The following records shall be retained for the duration of the FSV #1 Operating License.

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the FSAR,

b. Records of new and IRRADIATED FUEL inventory, fuel transfers and assembly burnup histories,

c. Records of radioactive shipments,

d. Records of facility radiation and contamination surveys,

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e. Records of radiation exposure for all individuals entering radiation control areas,

f. Records of gaseous and liquid radioactive material released to the environs,

g. Records of transients or operational cycles for those facility components identified in Table 5.6-1,

h. Records of training and qualification for current members of the plant staff,

i. Records of In-service Inspections performed pursuant to these Technical Specifications,

j. Records of Quality Assurance activities required by the QA Manual,

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59,

l. Records of meetings of the PORC and the NFSC,

m. Records of the service lives of all snubbers (required by Specification 3.7.10) including the date at which the service life commences and associated installation and maintenance records,

n. Records of secondary water sampling and water quality,

o. Records of reactor tests and experiments,

p. 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", and

q. Records and logs pertaining to the environmental monitoring program including baseline data from the pre-operational environmental monitoring programs (both radiological and non-radiological).

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6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 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.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 20.203(c)(5) of 10 CFR 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mR/h but equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates 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). Individuals qualified in radiation protection procedures (e.g., Health Physics personnel) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals 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, or

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, or

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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 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates 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. 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.

For individual areas accessible to personnel with radiation levels of greater than 1000 mR/h 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.

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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.

6.14 OFF-SITE 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);

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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.

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.

6.16 FUEL SURVEILLANCE PROGRAM

The Fuel Surveillance Program and any changes to the Program shall be approved by the Commission prior to implementation. This Program shall include provisions to submit results of the Fuel Surveillance Program examinations to the Commission in a timely fashion.

6.17 ENVIRONMENTAL QUALIFICATION

a. By no later than June 30, 1982, all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines; or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979, to the extent applicable to a gas cooled reactor. Copies of these documents are attached to Order for Modification, of License No. DPR-34 dated October 27, 1980.

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b. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588, to the extent applicable to a gas cooled reactor. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.18 INSERVICE INSPECTION AND TESTING PROGRAM

a. The Inservice Inspection and Testing Program shall be developed by PSC and approved by the Commission prior to implementation. This program will be implemented prior to plant operation in the POWER MODE after the fifth refueling outage.

b. Proposed changes to the Inservice Inspection and Testing Program shall be evaluated per the criteria of 10 CFR 50.59. Proposed changes involving an unreviewed safety question shall be submitted to the Commission for approval prior to implementation. Proposed changes not involving an unreviewed safety question may be implemented without prior Commission approval, but all such changes shall be reported to the Commission in the annual 10 CFR 50.59 summary report.

c. The Inservice Inspection and Testing Program shall be based on the applicable portions of ASME Boiler and Pressure Vessel Code Section XI, Divisions 1 or 2 as appropriate, for inspection and testing requirements, methodology, and acceptance criteria.

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.

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ATTACHMENT 2

TO P-88184

TECHNICAL SPECIFICATION UPGRADE PROGRAM

JUSTIFICATION OF CHANGES FROM PREVIOUS SUBMITTALS

TECHNICAL SPECIFICATION UPGRADE PROGRAM

JUSTIFICATION OF CHANGES FROM PREVIOUS SUBMITTALS

This attachment provides PSC's justification for the changes that have been made to the draft upgraded Technical Specifications since previously docketed submittals.

In Attachment 1, the revised Final Draft specifications have been margin-marked to identify recent changes. Spelling corrections, punctuation changes, capitalization of defined terms, and minor grammar corrections are not margin-marked, but all other revisions have been identified. In the Justifications, changes that are clear and of relatively minor consequence are not described and the reason for these changes is briefly described (editorial, clarification, correction, etc).

The previously docketed submittals cited in this attachment include the following:

PSC letter, Brey to Berkow, dated 2/20/87 (P-87063), Attachment 1

PSC letter, Brey to Calvo, dated 12/23/87 (P-87441), Attachments 1 and 2

PSC letter, Brey to Calvo, dated 2/2/88 (P-88045), Attachment 3

PSC letter, Brey to Calvo, dated 3/8/88 (P-88082), Attachment 3

PSC letter, Brey to Calvo, dated 2/8/88 (P-88025)

PSC letter, Brey to Calvo, dated 3/22/88 (P-88062)

SECTION 1.0, DEFINITIONS

Last docketed submittal: Pages 1, 5, 7 -- P-87441
Table 1.1 -- P-88045
Remainder -- P-87063

Changes and Justifications:

Definition 1.5 - editorial
Definition 1.9 - editorial
Definition 1.12 - clarification
Definition 1.16 - revised to more correctly reflect the way beta and gamma energies per disintegration are accounted for in the FSV E-BAR calculations. Also added clarifications for consistency with LCO 3.4.2
Definition 1.18 - editorial
Definition 1.29 - simplification, per discussion with NRC
Definition 1.30 - clarification
Definition 1.31 - clarification
Definition 1.32 - editorial
Definition 1.33 - editorial
Definition 1.38 - reference correction

Table 1.1, Note @ - As discussed with the NRC during a meeting on March 15 and 16, 1988, the # note applicability was changed to the RMS "Off" position and the @ note applicability was changed to the ISS "Low Power" position, as these are the only cases where they are needed. Also, the @ note was revised to allow power increases to 40% for surveillance or other testing, without it being considered a Mode change. This is required to provide operating flexibility and margin above the setpoints of many of the PPS channels while necessary testing is being performed.

Table 1.2 - Revised 'R' interval to "At least once per 18 months" for simplification and clarity for scheduling purposes. The intent is consistent with the previous "not to exceed 18 months", except that it is clearer that the requirements of 4.0.2 apply. This change is consistent with STS.

SECTION 2.0, SAFETY LIMITS

Last docketed submittal: Section 2.2.1 -- P-88025 and P-87063
Remainder -- P-87063

Changes and Justifications:

SL 2.1.1 - These Specifications were revised in a conservative direction to more accurately reflect the concern for fuel kernel migration and in a manner that is consistent with the way PSC has been implementing the requirement.

The concern addressed in this revision can most easily be understood by looking at the curve originally presented as Figure 2.1.1-1, the Core Safety Limit. This curve shows integrated operating time limits for various power to flow ratios, for example, 2 minutes at P/F of 2.5 and 100 hours at P/F of 1.17.

The previous draft Specifications would have allowed 2 minutes at P/F 2.5 <u>and</u> 100 hours at P/F of 1.17. Further review reveals that this is not the intent. Either condition by itself could result in undesirable fuel kernel migration.

The intent of the Safety Limit, as reflected in the revised draft is to limit power-to-flow transients so that their cumulative effect does not exceed the effect of any single transient that exceeds the curve of Figure 2.1.1-1 (previous draft). For each transient, the fraction of the allowable operating time is determined All of the transient fractions are then summed and limited to a total of 1.0.

LSSS 2.2.1 - Added ACTION b to allow 12 hours to adjust the setpoint of the Linear Channel - High Neutron Flux channel if required by a power level change. The detector decalibration curve, Figure 2.2.1-1, identifies different setpoints for different power levels. In the event of an unplanned power reduction from 80% to 65%, for example, a setpoint adjustment is required and 12 hours is allowed to make this change before considering the channels inoperable.

SL BASIS, page 2-12 - Editorial.

SL BASIS, page 2-15 - Added relief setpoint clarification.

SL BASIS, page 2-16 - Added relief device setpoint clarifications.

SECTION 3/4.0, APPLICABILITY

Last docketed submittal: Section 3.0.5 -- P-88045
Remainder -- P-87063

Changes and Justifications:

3.0.3 - As discussed during the March 15/16, 1988 meetings, changed requirement to be in at least LOW POWER in the first 12 hours vs. in STARTUP. This revision is discussed in the BASIS for 3.0.3, as defining an orderly shutdown at FSV. Reducing power to below 30% in 12 hours minimizes transient effects on the steam generators during boilout and allows for orderly re-orificing of core flows.

3.0.4 - As discussed during the March 15/16, 1988 meetings, this specification was revised per Generic Letter 87-09.

3.0.5 - Revised for clarity in describing how the CALCULATED BULK CORE COOLING concept is used to determine the applicability of various requirements. Also revised to clarify the calculational methodology.

4.0.3 - Revised per Generic Letter 87-09.

4.0.4 - Revised per Generic Letter 87-09.

4.0.5.a - Revised to delete ISIT Program Plan implementation information, as this is addressed in Specification 6.18.

BASIS page 3/4 0-7 - Revised per Generic Letter 87-09.

BASIS page 3/4 0-8 - Added per Generic Letter 87-09.

BASIS page 3/4 0-9 - First paragraph of 3.0.2 added per Generic Letter 87-09. Second paragraph added for clarification.

BASIS page 3/4 0-10 - Added per Generic Letter 87-09 except that the third paragraph was revised to allow a full 24 hours to be SHUTDOWN from any plant operating condition, based on the precedence of the current FSV Technical Specifications. GL 87-09 would only have allowed 12 hours if below 30% power.

BASIS page 3/4 0-11 - The first paragraph was added to explain the ACTION to reduce power to LOW POWER conditions in the first 12 hours, as discussed above. The second paragraph was added per Generic Letter 87-09.

BASIS page 3/4 0-12 - Paragraphs for 3.0.4 added per Generic Letter 87-09. Paragraph for 3.0.5 added for clarification.

BASIS page 3/4 0-13 - Clarification.

BASIS page 3/4 0-14 - Revised per Generic Letter 87-09.

BASIS page 3/4 0-15 - Deleted paragraph regarding previous 4.0.2 that was moved to Table 1.2.

BASIS page 3/4 0-16 - Added per Generic Letter 87-09.

BASIS page 3/4 0-17 - Revised per Generic Letter 87-09.

BASIS page 3/4 0-18 - The first revision is per Generic Letter 87-09. The revision to 4.0.5 deletes ISIT implementing information, as this is included in Specification 6.18.

SECTION 3/4.1, REACTIVITY CONTROL SYSTEMS

Last docketed submittal:

3/4 1.1 -- P-88045 (page 1, BASIS page 1 only)
3/4 1.5 -- P-88082
3/4 1.6 -- P-88045
Remainder -- P-87063

Changes and Justifications:

3.1.1, ACTION a - deleted Action for slack cable alarm as this has been covered by new ACTION i.

3.1.1, ACTION d.2 - Added provision that up to 4 control rod motor temperatures can exceed 250 degrees F. This allows these four control rod pairs to be considered OPERABLE for determining the SHUTDOWN MARGIN, provided each rod has an extrapolated scram time less than 152 seconds, as determined on a daily basis. As explained in the BASIS, daily testing demonstrates reliable operation and scram capability. Four control rods are allowed greater than 250 degrees F as an upper limit that supports reasonable plant operation while controlling general CRDM temperature elevation.

3.1.1, ACTION h.2 - Editorial.

3.1.1, ACTION i, Added to address slack cable condition. In the event of a slack cable alarm, procedures specify attempting to clear the alarm by manipulating the control rod pair a few inches. If the alarm clears (due to tightening the cable) but the control rod pair does not move, the rod is immovable. Action a applies as soon as that determination is made, as it does for any other condition that results in an immovable control rod pair. Action i addresses the activities to be taken if the slack cable alarm is due to some other condition, as further explained in the BASIS.

4.1.1.d - Revised partial scram test for control rods that are being withdrawn during operations after initial withdrawals to achieve criticality. A withdrawal of only 10 inches or so at the full insertion of the CRD does not produce meaningful and trendable extrapolated scram times. Initial testing prior to criticality can be performed from greater rod heights but once the reactor is operating, the regulating rod must compensate for these longer partial scram tests. To avoid unnecessary cycling of the regulating rod and to avoid any single rod withdrawal prohibits, this SR has been revised to demonstrate freedom of movement and scrammability. Within 7 days, partial scram testing will be performed to obtain an extrapolated scram time.

4.1.1.g.3 - Editorial.

BASIS page 3/4 1-6 - Editorial.

BASIS page 3/4 1-7 - Editorial clarification of extrapolated scram time, added discussion that four control rod pairs may exceed a CRDM temperature of 250 degrees F.

BASIS page 3/4 1-8 - Editorial clarifications. Added clarification that instrumentation failure could be a cause of a slack cable alarm.

BASIS page 3/4 1-9 - Enhanced discussion on slack cable alarm Actions. Editorial clarification of surveillances.

BASIS page 3/4 1-10 - Clarification of partial scram tests as discussed above.

Table 3.1.2-1, page 3/4 1-13 - Clarified that Rod-in Limit Switch OPERABILITY can be determined from the last tested condition.

BASIS page 3/4 1-19 - Editorial.

BASIS page 3/4 1-20 - Editorial.

BASIS page 3/4 1-26 - Editorial.

3.1.4.1.a - Revised conditions during which control rod pairs are manipulated in groups. Added manipulation of up to six control rod pairs to the exception list, as permitted in Specification b.2.

3.1.4.1.b - Added clarification that partial scram testing requires control rod pairs to be moved from their fully withdrawn or fully inserted positions.

3.1.4.1 Action a.2 - Revised to reflect orderly shutdown, consistent with Specification 3.0.3.

BASIS page 3/4 1-30 - Added BASIS for maximum control rod pair worth limitation. Other editorial revision.

BASIS page 3/4 1-31 - Editori[illegible]lso discussed orderly power reduction Action.

3.1.4.2 - Editorial.

BASIS page 3/4 1-35 - Clarified rod-in indication that may be used for control rod pair position verification. Normally, the rod-in limit indication (green light) verifies control rod pair insertion. If this is inoperable, the analog or digital indications may also be used. The use of these other indications is acceptable, as a watt-meter test is required within 12 hours by Specification 3.1.2.2 in the event rod-in limit indication is not available.

BASIS page 3/4 1-39 - Clarification.

4.1.6.1.e - Added reference to surveillance for testing RSD material after entry of condensed water, consistent with 4.1.6.2. Water entry into an RSD hopper is a remote possibility in all operating modes.

3.1.6.2 - Editorial.

4.1.6.2.a, b - Revised SRs to ensure RSD operability on control rod pairs capable of being withdrawn, consistent with the LCO applicability.

4.1.6.2.c - Added annual calibration of gas pressure instrumentation, consistent with 4.1.6.1. This annual surveillance is applicable in various operating conditions.

4.1.6.2.d - Editorial.

4.1.6.2.e - Editorial.

BASIS page 3/4 1-44 - Clarified contribution of Sm-149. Corrected RSD worths.

BASIS page 3/4 1-46 - Added [illegible]tion justification.

4.1.7 - Editorial.

BASIS page 3/4 1-48 - Editorial.

SECTION 3/4.2, CORE IRRADIATION, TEMPERATURE AND FLOW LIMITS

Last docketed submittal: 3/4.2.1 - P-88045
3/4.2.4 - P-88082
Remainder - P-87063

Changes and Justification:

3.2.1, ACTION - Editorial, for consistency with LCO.

4.2.1 - Editorial, for consistency with LCO.

BASIS page 3/4 2-2 - Editorial change and deleted itemization of loads and stresses as being overly detailed.

Figure 3.2.2-1 - Revised to include all expected combinations of operating parameters. Existing curves were extrapolated, making use of the same methodologies and including the same uncertainties used during their initial development. Shading is provided to preclude operation with outlet temperatures above 1555 degrees F. This revision will be addressed as a "C" category comment.

BASIS page 3/4 2-6 - Clarification of description of transverse flow.

BASIS page 3/4 2-7 - Editorial.

3.2.3 - Deleted LOW POWER from Applicability for consistency with the surveillances. Both SR 4.2.3 in the TSUP and SR 5.1.7 in the existing FSV Technical Specifications only determine RPF discrepancies above 30% power. This revision is consistent with the current requirements for use of COMPARISON REGIONS.

4.2.3 - Clarification and consistency with Applicability.

4.2.3.b - Editorial.

BASIS page 3/4 2-10 - Editorial.

BASIS page 3/4 2-11 - Clarification.

3.2.4 ACTION c - Added REFUELING** as it is possible, although unlikely, to have a CBCT above 760 °F in this mode.

BASIS page 3/4 2-21 - Editorial.

3.2.6 - Significantly revised as described in the discussion of changes to SL 2.1.1. Other editorial changes.

SECTION 3/4.3.1, PLANT PROTECTIVE SYSTEM

Last docketed submittal: Pages 3/4 3-1 through 15 -- P-88025
Pages 3/4 3-16 through 28 -- Existing License
Pages 3/4-29 through 36 -- P-88025
Page 3/4-37 -- Existing License

Changes and Justifications:

3.3.1, Applicability, ACTION 3.3.1.a - Revised to be consistent with the standard Technical Specification (STS) format. The overall intent of these sections has not changed. This revision provides clarity and consistency in format with the TSUP.

ACTION 3.3.1.b.1 - Revised ACTION time from 12 hours to 24 hours. The 12 hours in the existing Technical Specifications to shut down when the minimum number of OPERABLE channels is not met is overly restrictive as the LWR STS philosophy in similar actions requires an orderly shutdown (i.e., 24 hours at FSV).

Table 3.3.1-1, Item 4 - LCO 4.9.2 in the existing license was a "one time" Dew Point Moisture Monitor test which injected moisture-laden gas into primary coolant. The test has been completed with favorable results. Therefore, this LCO is eliminated and will be discussed with the "C" Category comments.

Table 3.3.1-3 (Part 2), Page 3/4 3-10 - Clarification.

ACTION statements for 10a, 10b, 10c, and 10d - The applicable ACTION statements for SLRDIS valves are included in 3/4.7.8.

Note d, Page 14 - Editorial.

Note f, Page 14 - As noted above, LCO 4.9.2 in the existing license was a "one time" test which has been completed with favorable results. Therefore, this LCO is eliminated.

Note h, Page 14 - Editorial.

Note i, Page 14 - Editorial.

Note k, Page 15 - Editorial.

Table 4.3.1-3, Page 26 - Editorial.

BASIS - Editorial.

SECTION 3/4.3.2, MONITORING INSTRUMENTATION and

SECTION 3/4.3.3, THREE ROOM CONTROL COMPLEX TEMPERATURE MONITORING

Last docketed submittal: 3/4.3.2.3 -- P-88045
Remainder -- P-87063

Changes and Justifications:

3.3.2.1.a - Editorial.

3.3.2.1.b - Editorial.

ACTIONS 3.3.2.1.a.2 and 3.3.2.1.b - Changed the acceptable alternate operating modes by deleting POWER. This mode can not be achieved without passing through LOW POWER operation. Therefore, this change is a clarification and a simplification.

Footnote, Page 38 - Clarification.

4.3.2.1 - Editorial.

4.3.2.1.c - Editorial.

4.3.2.1.d - Editorial.

4.3.2.1.e - Clarification.

ACTION 3.3.2.2.a - Editorial.

ACTION 3.3.2.2.b - Editorial.

4.3.2.7 - Editorial.

3.3.3, 4.3.3 - Changed limit from "shall be less than 115 degrees F" to "shall not to exceed 115 degrees F". This is consistent with the LWR STS and accounts for periods when the temperature is 115 degrees F.

Bases, general - Clarifications and editorial.

SECTION 3/4.4, PRIMARY COOLANT SYSTEM

Last docketed submittal: 3/4.4.1 -- P-87441
Remainder -- P-87063

Changes and Justification:

3.4.1.1 - Editorial.

3.4.1.2, ACTIONS - Editorial.

BASIS page 3/4 4-7 - Clarification.

BASIS page 3/4 4-8 - Clarification and editorial correction.

3.4.2.d - Clarification that Sr-90 plateout limit is bone dose equivalent, per current FSV Technical Specification 4.2.8.d.

4.4.2.1 - Clarified that daily surveillance verifies gross gaseous activity level.

4.4.2.1.b - Deleted requirement to analyze a grab sample if the gross activity recorder is inoperable. This revision is in agreement with the current FSV Specification SR 5.2.11. PSC had originally proposed the addition of the recorder for completeness, but this appears excessive considering the monitoring and alarm functions of the monitor.

4.4.2.2 - Clarified SR for gaseous and plateout activity levels.

4.4.2.2.a - Relocated surveillance for sensitivity of gross activity monitor to 4.4.2.3. Also, clarified that circulating iodine and plateout Sr-90 activity levels can only be estimated from the grab sample analysis.

4.4.2.2.b - Editorial.

4.4.2.2.c - Clarified plateout probe use to determine circulating iodine and plateout Sr-90 levels.

4.4.2.3 - Clarified surveillances for gross activity monitor, consistent with LWR post accident primary coolant radiation monitors.

NOTE 1, page 3/4 4-11 - Revised Note 1.b consistent with E-BAR definition.

BASIS page 3/4 4-12 - Editorial clarifications.

BASIS page 3/4 4-13 - Editorial consistency.

BASIS page 3/4 4-14 - Clarified discussions on grab sample analysis. BASIS page 3/4 4-16 - Clarified gross activity monitor surveillance.

3.4.3 Action d - Revised Actions after exceeding the 1000 ppm-day limit to reduce further operating time in a limited acceptable condition. This revision allows an orderly power reduction.

BASIS page 3/4 4-20 - Deleted CO Infrared Analyzer as it is not normally in service and is not relied upon in the FSAR. Also made editorial correction.

BASIS page 3/4 4-26 - Revised statement regarding conservatism of 90-day ACTION for clarity. Deleted reference to CO Infrared Analyzer as it is not relied upon in the FSAR. Deleted paragraph regarding exposure of reserve shutdown material to primary coolant. The RSD material is not exposed to primary coolant during normal plant operations and therefore the effects of primary coolant impurities on the boron-carbide balls is not a BASIS for the impurity restrictions.

SECTION 3/4.5, SAFE SHUTDOWN COOLING SYSTEMS

Last docketed submittal: 3/4.5.2.1 - P-88082
3/4.5.5 - P-88082
Remainder - P-87441

Changes and Justification:

BASIS page 3/4 5-4 - Editorial.

BASIS page 3/4 5-5 - Clarified that 12 hours to be in SHUTDOWN is a reasonable time, considering the change to 3.0.3. Also, other editorial changes.

3.5.2.1 - Added component designators for clarity.

3.5.2.2 - Added component designators for clarity.

BASIS page 3/4 5-9 - Deleted statement that backup bearing water is available because this is not true at all times. Plant procedures do not provide this feature at lower power levels.

3.5.3.1 - Deleted ACTION b allowing power reduction below 35% as the capability of liner cooldown is not relied upon after a power reduction from operating levels where there is significant decay heat.

BASIS page 3/4 5-15 - Clarification.

BASIS page 3/4 5-16 - Deleted paragraph addressing previous ACTION b, other editorial correction.

BASIS page 3/4 5-18 - Clarification.

BASIS page 3/4 5-21 - Editorial.

3.5.5, ACTIONS - Editorial.

BASIS page 3/4 5-27 - Editorial.

SECTION 3/4.6, PCRV AND CONFINEMENT SYSTEMS

Last docketed submittal: 3/4.6.2.1 -- P-88082
3/4.6.2.2 -- P-87441
3/4.6.3 -- P-37441
3/4.6.4 -- P-88082
3/4.6.5.2 -- P-88082
3/4.6.5.3 -- P-88082
Remainder -- P-87063

Changes and Justifications:

3.6.1.1.e - Changed acceptable pilot stage bellows pressure to 200 psig, consistent with current instrumentation. As stated in the BASIS, the actual setpoint is arbitrary as pressure would only be experienced during testing, and a range less than 200 psig is sufficient to detect bellows leakage.

4.6.1.1.b - Revised to clarify use of ASME Code.

BASIS page 3/4 6-3 - Revised to reflect bellows alarm pressure, as stated above.

4.6.1.2 - Revised to clarify use of ASME Code.

BASIS page 3/4 6-7 - Editorial.

BASIS page 3/4 6-9 - Editorial.

3.6.2.1 - Revised tube temperature limits, added Action d, revised SR 4.6.2.b, and revised BASIS to reflect average tube temperatures in the LCS. See Justification at the conclusion of this Attachment.

3.6.2.1 Action a - Revised actions to address control rod movements, consistent with previous discussions. This corrects an oversight in the P-88082 submittal.

4.6.2.1.e.2 - Editorial.

3.6.2.2 - Editorial.

4.6.2.2 - Editorial.

3.6.2.2 footnote, page 3/4 6-21 - Corrected to be specific to sidewall tubes and to require a 250 degree F limitation. This is consistent with the actual FSV condition and with the concrete design criteria.

BASIS page 3/4 6-22 - Editorial, other changes per above tube temperature discussion.

BASIS page 3/4 6-23 - See above discussion on tube temperatures.

BASIS page 3/4 6-24 - Editorial.

BASIS page 3/4 6-27 - Editorial.

4.6.4.1 - Revised to reflect annual tendon surveillance program, per P-87234.

3.6.4.2 - Revised discussion of primary coolant boundary versus primary coolant pressure boundary, consistent with the HTGR terminology in ANSI 58.4.

3.6.4.3 - Added LCO, Action and Surveillances to include isolation valves as included in current FSV ISI Tech Spec SR 5.2.24.

4.6.4.3.d - Revised frequency to 'once per 18 months', as these are all surveillances that are done during a refueling shutdown, and refueling has been clarified as at least once per 18 months.

4.6.4.3.e - Editorial.

BASIS pages 3/4 6-40 to 6-44 - Revisions consistent with the above discussions.

4.6.5.1.b, c - Editorial clarifications.

BASIS page 3/4 6-51 - Editorial.

BASIS page 3/4 6-54 - Editorial clarifications and added supporting justifications for the requirement for 70 OPERABLE louver panels.

Justification for Average PCRV Liner Cooling System Temperature Limits in TSUP 3/4.6.2.

The design criteria for the PCRV, DC-11-1, required that loads that produce stress and/or strain in the PCRV be considered in the design so that the overall structural response of the PCRV would be essentially elastic. The basis for the design loads and load combinations is described in FSAR Section E.1.2.4.

Regarding temperature loads, FSAR Section E.1.2.4 states that the PCRV experiences non-uniform temperatures due to the transfer of heat from the cavity and penetration liners to the colder outer surfaces of the concrete. These non-uniform temperatures result in thermal stresses in the self-strained structure. In concrete, these stresses tend to be relaxed by creep and other inelastic effects, particularly in areas of local stress concentration. Because of this, only the bulk temperature distribution needs to be considered in establishing the thermal loading of the PCRV.

For the purpose of analysis, an "effective temperature" was used. The "effective temperature" is the temperature obtained by projecting the temperature gradient in the bulk of the concrete to the liner concrete interface. The "effective temperature" used in the PCRV design is up to 130°F per DC 11-1.

The thermal load in the PCRV has been determined using this "effective temperature" on the liner and the appropriate ambient boundary conditions on the outside surface of the concrete.

In addition to temperature variation through the walls, the temperature of the concrete varies locally between cooling tubes involving only a small amount of concrete. The following limits were set on local maximum concrete temperatures at the liner-concrete interface.

a) Between Tubes.

With Both Cooling Circuits in Operation up to 150°F.

b) Adjacent to Members that Penetrate the Thermal Barrier and Adjacent to Joints in the Thermal Barrier.

With All Cooling Tubes Operating	200°F
Adjacent to a Non-operating Cooling Tube	250°F

During startup testing, liner cooling tube temperature rise and/or concrete temperatures exceeding the design criteria were identified at seven "hot-spot" locations. These are discussed in FSAR Section 5.2.9.8.

The liner hot spot areas are generally caused by a discontinuity in the thermal barrier, providing a lower resistance path for the heat to the liner concrete interface. Structural attachments to the liner also provide a heat conduction path to the liner. Higher than expected heat fluxes can also be caused by bypass or impinging helium flow patterns, or by unsuitable cooling tube configurations.

The severity of these seven hot spots encompasses liner cooling tube temperature rises up to 47°F and local concrete temperatures up to 354°F, as shown in Table 1.

In 1981, the NRC contracted with Los Alamos National Laboratory (LANL) to undertake detailed analyses of the PCRV hot spot phenomenon based upon FSV operating history. The objective of this study, described in FSAR Section 5.9.2.8.9, was to calculate the thermally-induced stresses in the regions of the PCRV where concrete temperatures exceed FSAR hot spot criteria and to compare the calculated thermal stresses to ASME and FSAR limiting values.

The four most severe hot spots in the PCRV were analyzed for thermal stresses. These hot spots were all characterized by concrete temperatures in excess of 300°F. Generally, the hot spot regions were not structural members, but one of the hot spots, the core support floor, was in a region where structural load bearing capacity was an important consideration.

In all of the calculations, the thermally generated stresses caused by increasing the concrete temperature some 150°F over the hot spot temperature limit were well below the ultimate stress for the existing state of triaxial stress and temperature, as defined in ASME codes and the FSAR.

Since the temperatures in the three unanalyzed hot spots were smaller, it was concluded that the stresses in those cases would be less severe than the cases calculated. LANL concluded that since these stresses are all within the elastic limits, more complicated thermal loading histories, such as cyclic cooling and heating, should have negligible effect.

During normal power operations, other localized variations in individual liner cooling tube temperatures have been observed. These variations may be attributed to variations in heat flux or variations in cooling water flow patterns. Based on comparison with the bounding conditions established by the analysis of the hot spots discussed above, liner cooling tube temperature rises of up to 25°F would result in stress less severe than those calculated by LANL. Since LANL concluded that hot spots with lower stress are acceptable without further analysis, then other tube temperature rises up to 25° F are considered acceptable without further analysis.

The limitation on overall liner cooling system temperature rise of 20°F (LCO 3.6.3) continues to be in effect and ensures that the overall bulk temperature of the PCRV remains within design criteria.

TABLE 1

PCRV LINER HOT SPOTS

	Liner Cooling Tube ΔT**	Local Maximum Concrete Temperature at the Liner/Concrete Interface - °F**	Effective Concrete Temperature at the Liner/Concrete Interface - °F**
1) Top Head Penetrations	20°F @ 55% 30°F @ 100%	201°F @ 100%	128°F @ 100%
2) Core Outlet Thermocouple Penetrations*	20°F @ 80% 25°F @ 100%	200°F @ 20% 346°F @ 100%	120°F @ 100%
3) Core Barrel Seal Area*	40°F @ 60% 47°F @ 100%	250°F @ 27% 326°F @ 100%	120°F @ 100%
4) Peripheral Seal Area	20°F @ 40% 28°F @ 100%	200°F @ 100%	120°F @ 100%
5) Loop Divider Baffle Area*	15°F @ 100%	200°F @ 20% 354°F @ 100%	120°F @ 100%
6) Steam Generator Penetrations	16°F @ 100%	200°F @ 40% 220°F @ 100%	120°F @ 100%
7) Cross Over Pipe*	3°F @ 100%	200°F @ 11% 300°F @ 100%	110°F @ 100%
FSAR	20°F	150 Normal 200 Normal Hot Spot 250 Non-operating tube	130°F

* Subject of detailed analysis by LANL
** Temperatures are presented at various reactor power levels.

SECTION 3/4.7, PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

Last docketed submittal: 3/4.7.1.1 -- P-87441
3/4.7.1.5 -- P-87441
3/4.7.1.6 -- P-87441
3/4.7.1.7 -- P-87441
3/4.7.2 -- P-88045
3/4.7.4 -- P-88082
3/4.7.5 -- P-87441
3/4.7.6.3 -- P-88045
3/4.7.6.4 -- P-88082
3/4.7.8 -- P-88082
Remainder -- P-87063

Changes and Justifications:

3.7.1.1.b - Revised requirement to have an OPERABLE auxiliary boiler in the event the motor-driven feed pump is not OPERABLE. Previous specification required the auxiliary boiler to be operating. This change is based on the recent HELB analysis that indicated that the most limiting time for restart of forced circulation following DBA-2 is 60 minutes (FSAR 14.11.2.2). The OPERABLE auxiliary boilers can be brought on line in 60 minutes in support of this requirement. This position is also reflected in the BASIS, page 3/4 7-2. This change is considered a 'C' category comment and will be included in the justifications for these comments.

3.7.1.1, Applicability - Added condition in the footnote that the specification is not applicable with the PCRV pressure below 100 psia. The boiler feed pumps are required for a depressurization accident from full power. When the PCRV is already depressurized, the reactor power is low and the consequences of a depressurization accident are substantially reduced to where the flows provided by these pumps are not required. This position is reflected in the BASIS, page 3/4 7-3.

4.7.1.1 - Editorial.

BASIS page 3/4 7-2 - Editorial clarifications, other changes discussed above. Also, added clarification that auxiliary boilers are not operated above 65% reactor power.

BASIS page 3/4 7-3 - Discussed above.

BASIS page 3/4 7-6 - Editorial.

BASIS page 3/4 7-7 - Clarification that FSV loop dumps associated with steam generator tube leaks have historically been manual.

4.7.1.3 - Identified the surveillance requirements as the ISIT Program Plan referenced in Specification 4.0.5 will not be implemented concurrent with the TSUP.

BASIS page 3/4 7-9 - Added BASIS for surveillance.

3.7.1.4 - Deleted "dose equivalent" from I-131 requirement, consistent with current FSV Technical Specification SR 5.3.7. Also made this change in the Action and in 4.7.1.4.

3.7.1.5 and 3.7.1.6 - Changes in both specifications were made to allow setpoint adjustments to be made with system steam conditions as close to the setpoint as possible. These conditions occur around 50% power, rather than 30% as previously submitted. Also, 7 days are allowed for this setpoint verification as safety functions are not significantly compromised. Setpoint verification is accomplished as follows: the valve body and disc temperatures must be close to those expected near the setpoint, as thermal expansion can affect clearances and other critical dimensions in the valves. Once these conditions are reached in the piping, a test device is connected that applies an external mechanical force on the valve stem assembly. This test device applies a force that simulates steam pressure at the setpoint and the valve is adjusted to lift. The test device contains oil whose temperature must be controlled, occasionally by removing the device to a cooler area which extends the time required. When maintenance activities require valve disassembly during cool conditions, the stem position is marked at the previous setting and this position is restored after reassembly; this activity is referred to in 3.7.1.6 as estimating the setpoint.

3.7.1.5, Action a.1 - Editorial.

4.7.1.7 - Provided pump surveillance requirements as the 4.0.5 ISIT program plan will not be implemented with the TSUP.

4.7.3 - Revised minimum instrument air receiver pressure to 85 psig. This reflects system design as the third, standby instrument air compressor starts at 85 psig. If pressure drops below this point, there is a problem and the system should be considered inoperable. Also revised BASIS page 3/4 7-23 to reflect this position.

3.7.4.1, Action d - Editorial.

3.7.4.1, Action e - Deleted previous Action e as it provided Actions with backup cooling and no firewater pumps OPERABLE. This is not meaningful as the firewater pumps are also required for the identified backup. Also reflected this in the BASIS, page 3/4 7-30.

3.7.4.2, Action - Revised to allow alternate cooling as needed during extended service water outages (e.g., during tower maintenance). Action b revised for consistency with loss of firewater Actions in 3.5.5.

3.7.5, Applicability - Added all cases with a CBCT greater than 760 degrees F, as depressurization actions can be required in all these modes.

3.7.5, Action c.2 - Revised to reflect requirements of previous actions with single inoperable trains.

4.7.5.c - Added SR for purification cooling water pumps, consistent with current FSV specification SR 5.2.24.g.

BASIS page 3/4 7-33 - Editorial clarification.

BASIS page 3/4 7-43 - Editorial.

3.7.8 - Editorial.

BASIS page 3/4 7-57 - Added BASIS for surveillance.

4.7.9.a - Editorial.

4.7.9.b.2 - Revised methyl iodine penetration criteria from 5% to 3%, per PSC letter, Brey to Calvo, dated 5/15/87 (P-87133), regarding NUREG-0737 technical specifications.

4.7.9.c - Revised frequency to after every 720 hours of operation, consistent with the STS. PSC is adding elapsed time instrumentation to permit this frequency, subsequent to our previous position on this issue.

4.7.9.d.1,2 - Added per PSC letter P-87133 regarding NUREG-0737 technical specifications.

BASIS page 3/4 7-70 - Revised discussion supporting Applicability of the snubber specification above 5% power to reflect recent analysis documented in FSAR Section D.4.2. This analysis indicates that snubbers are not required below 8% power.

SECTION 3/4.8, AUXILIARY ELECTRIC POWER SYSTEMS

Last docketed submittal: P-88062

Changes and Justifications:

3.8.1.1.b.2 - Clarification.

ACTION 3.8.1.1 - Editorial.

4.8.1.1.2.a.8 - Editorial.

4.8.1.1.2.c.2 - Editorial.

4.8.1.1.2.e.5.d and 4.8.1.1.2.e.9 - Relocated surveillance per agreement in meeting with NRC on 3/15/88.

4.8.4.c - Editorial.

SECTION 3/4.9, FUEL HANDLING AND STORAGE SYSTEMS

Last docketed submittal: 3/4.9.1 -- P-88045
3/4.9.2 -- P-88045
Remainder -- P-87063

Changes and Justifications:

3.9.1 - Editorial.

3.9.2, 4.9.2 - Editorial.

BASIS page 3/4 9-4 - Added clarification of requirements for reactor internal maintenance, when no CORE ALTERATIONS are involved.

3.9.3 - footnote - Editorial.

4.9.3.3 - Revised daily visual inspection of the crane to once per 14 days. The crane is inaccessible for inspection while in use and operation often involves multiple-day events. The daily inspections were based on earlier plant procedures but have since been deemed impractical. A 14 day inspection is an appropriate extension of the load tests and other examinations that are a part of PSC's OSHA program to ensure safe crane operations.

4.9.3.3 - Other editorial changes.

BASIS page 3/4 9-9 - Editorial.

3.9.4 - Clarification that concern is for fuel element surface temperature.

4.9.4 - Deleted instrumentation SRs, consistent with TSUP philosophy. Considered C category change.

BASIS page 3/4 9-12 - Deleted reference to instrumentation.

3.9.6 - Added footnote that communications between the operations control room and the FHM control room are not required during control rod pair withdrawal for testing, when no other CORE ALTERATIONS are in progress, as this activity is closely monitored and sufficient controls and indications are available.

BASIS page 3/4 9-16 - Added explanation of footnote.

SECTION 3/4.10 - SPECIAL TEST EXCEPTIONS

Last docketed submittal: P-88082

Changes and Justifications:

BASIS page 3/4 10-2 - Revised FSAR reference to more appropriately cite maximum worth rod pair criteria.

SECTION 5.0, DESIGN FEATURES

Last docketed submittal: P-87063

Changes and Justifications:

5.1.1 - Added clarification about extent of evacuation procedure.

Figure 5.1-1 - Revised to reflect effluent release points.

5.2.1 - Revised to clarify primary and secondary containment analogies.

5.3.3 - Added clarification that not all eight test fuel elements currently reside in the core.

5.3.6 - Deleted overly specific description of control rod absorber bodies, as new absorber bodies may be different from these descriptions. Also, deleted description of rod pair withdrawal to 4 inches above the active core, to eliminate any confusion regarding whether internal core components are considered the top of the core.

5.3.6 - Clarification.

5.3.7 - Consistency with FSAR and Specification 3.1.5.

5.5 - Editorial.

5.6.1 - Editorial.

Table 5.6-1 - Clarification.

SECTION 6.0, ADMINISTRATIVE CONTROLS

Last docketed submittal: Pages 6-1, 6-2 -- P-88082
Pages 6-9 thru 6-21 -- P-88082
6.9.1.2 -- P-88045
6.12 -- P-88082
6.16, 6.17, 6.18 -- P-88082
Remainder -- P-87063

Changes and Justification:

Table of Contents - Editorial.

6.2 - Revised to delete organization charts per Generic Letter 88-06.

6.2.2.g - Revised to require procedural control of excessive overtime, without monthly Station Manager review. This is consistent with the current FSV operations, per current Technical Specification 7.1.1.2.i.

6.2.2.h - Revised per Generic Letter 88-06.

6.2.3 - Editorial.

6.5.1.6.i, j - Editorial.

6.5.1.6.k - Revised PORC responsibilities to delete review of unplanned releases, consistent with the latest version of the STS Administrative Controls provided by the NRC. Any unplanned releases that exceed the criteria for REPORTABLE EVENTS will be reviewed per 6.5.1.6.f.

6.5.1.7.d - Revised requirement for PORC to notify the Manager, Nuclear Production of disagreements between PORC and the Station Manager. Due to recent FSV organizational changes, the Station Manager is also the Manager, Nuclear Production. Consistent with the latest version of the STS provided by the NRC, the notification was changed to Vice President, Nuclear Operations.

6.5.2.4 - Editorial.

6.5.2.8 - Deleted previous paragraph i as this dealt with approval of the base reactivity curve which is more correctly dealt with in 6.5.2.9.b.

6.5.2.9 - Revised b for clarity and added d to reflect requirements of 3.10.1.

6.5.2.10 - Editorial.

6.5.2.12 - Revised to reflect FSV practice.

6.7.c - Revised to reflect notification of the Chairman, NFSC, consistent with current practice, per current Technical Specification 7.2.b.

6.8.1.e - Editorial.

6.8.2 - Revised to reflect current FSV practice for PORC review of post-maintenance tests. This is not addressed in the current FSV Specifications.

6.9.1 - Editorial.

6.9.1.1.c - Revised Startup Report section to delete quarterly status reports. The extended status of the startup test program at FSV leads to an indefinite number of quarterly reports that provide no new information. PSC proposes to submit a report upon completion of the Startup Test Program.

6.9.1.2 - Editorial.

6.9.1.2.d - Clarified that the annual diesel generator reliability report is required for the Standby Diesel Generators as they are subject to the reliability program in Specification 3/4.8.1.

6.9.1.3 - Editorial clarification.

6.9.3.d - Editorial clarification.

6.12.1 - Revised High Radiation Area definition to specify radiation greater than 100 mR/hr. This is consistent with the definitions in 10 CFR 20.202, as currently implemented in PSC's procedures, and is considered a clarification.