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August 5, 1988
Fort St. Vrain
Unit No. 1
P-88287

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 (TSUP) Revisions to Final Draft

REFERENCES:
1) PSC letter, Brey to Calvo, dated 5/27/88 (P-88184)
2) PSC letter, Brey to Calvo, dated 6/14/88 (P-88205)
3) NRC memorandum, Heitner to Calvo, dated 7/26/88 (G-88292)

Dear Mr. Calvo:

Attached are revisions to the Fort St. Vrain Technical Specification Upgrade Program (TSUP) draft specifications that were previously submitted in References 1 and 2. These revisions provide further clarifications and corrections as discussed with the NRC on July 13, 1988 (Reference 3). Included with the attachment is a tabulation of each affected specification and a brief description of the change.

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If you have any questions regarding this information, 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

Attachment

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

Mr. R. E. Farrell
Senior Resident Inspector
Fort St. Vrain

ATTACHMENT TO

P-88287

As a result of a telephone conference with the NRC on July 13, 1988, the following changes have been made to the FSV TSUP draft submittals dated May 25, 1988 and June 13, 1988: (the asterisked margin marks distinguish the most recent revisions).

Specification	Description of Change
SL 2.1.1 BASIS	Editorial clarification; revised subparagraph #3 to indicate that actual time periods are determined for each P/F RATIO interval during which the limits are exceeded.
LCO 3.1.1, Action i	Added Action to declare control rod inoperable if slack cable alarm cannot be fixed within 24 hours, to avoid entry into Specification 3.0.3 and shutdown.
LCO 3.1.6 BASIS	Revised reserve shutdown system discussion to reflect maximum temperature defect.
LCO 3.2.6, Action c	Editorial clarification; revised to indicate that the fraction of allowable operating time is determined for each P/F RATIO interval experienced during the transient.
Figure 3.2.6-2	Revised to show 100 hour allowable operating time with P/F of 1.05 to 1.17, consistent with the requirements of current SL 3.1.3.4.
LCO 3.2.6 BASIS	Editorial clarification; revised consistent with the LCO.
Table 3.3.1-1	Corrected Table 2.2.1-1 reference.
LCO 3.3.1 BASIS	Deleted discussion on SLRDIS valves, as this is contained in TSUP specification 3.7.8.
LCO 3.3.2.1	Editorial clarification; revised "alternate" to "alternately".
LCO 3.3.2.2 BASIS	Editorial clarification in header; changed LCO 3.3.2 to LCO 3.3.2.2.
SR 4.3.2.5.1	Editorial correction; changed Table 3.3.2.4 to Table 3.3.2-4.
SR 4.4.2.2	Editorial clarification; revised Sr-90 surveillance to determine bone dose equivalent activity, consistent with the LCO.

Specification	Description of Change
LCO 3.4.2 BASIS	Editorial clarification; revised Basis to discuss bone dose equivalent Sr-90 activity.
LCO 3.4.3 BASIS	Corrected discussions about carbon transport and weight loss allowances from "per year" to "per cycle". This reflects the original intent of the requirements and takes into account the fact that fuel cycles have historically not been equivalent to an annual cycle.
LCO 3.7.1.5	Added action to avoid Specification 3.0.3 shutdown requirements in event of inoperability of 2 EES safety valves.
LCO 3.7.1.5 BASIS	Revision to clarify boiler feed pump capacity.
SR 4.8.1.1.2.e.3	Clarified that load rejection test involves rejection of single largest load in lieu of 202 kW, in the event that the largest load is not actually 202 kW.
LCO 3.8.1 BASIS	Identified that single largest load is circulating water pump.
Table 4.8.4-1	Editorial clarification; abbreviations GTE and LTE were replaced with easier to read symbols for "greater than or equal to" and "less than or equal to", respectively.
LCO 3.9.6	Revised applicability to when CORE ALTERATIONS are conducted from the refueling floor, to clarify that communications are not required between the control room and the FHM control room when control rods are being withdrawn from the control room, as during normal operations. Also, deleted footnote reference from the surveillance, for consistency.
LCO 3.9.6 BASIS	Revised for consistency with applicability change.
DF 5.3.4	Added temperature coefficient concerns as reload requirements.
AC 6.5.1.6	PSC agreed to consider reinstating the requirement for PORC to review unplanned radiological releases to the environs, with some reasonable threshold value. Subsequently, PSC has found examples of several recent plant Technical Specifications that do not include this requirement (River Bend 1, Grand Gulf 1, Nine Mile Point 2, and Palo Verde 1). Based on these examples, PSC proposes that AC 6.5.1.6 remain as submitted in the May 25, 1988 draft.

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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 for 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 any P/F RATIO transient, the actual transient time period for each P/F RATIO interval during which the limit of Figure 3.2.6-1 is exceeded, 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.

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 or declare the affected control rod pair inoperable and comply with the requirements of Action b. *

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.

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BASIS FOR SPECIFICATION LCO 3.1.6/SR 4.1.6

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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 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. Based on calculated excess reactivity data in Table 3.5-4 and Section 3.5.3 of the FSAR, the maximum allowable temperature defect is 0.065 delta k, per LCO 3.1.5. Full Xenon decay is worth 0.032 delta k, per FSAR Table 3.5-4. Sm-149 build-up and 2 weeks of Pa-233 decay are worth about 0.007 delta k, and this value increases to about 0.024 delta k over several months, including full Pa decay. Based on the above, the total reactivity increase for 2 weeks after shutdown is 0.104 delta k. Therefore, reactor shutdown is assured for at least 2 weeks using only the reserve shutdown system, in the unlikely event that all control rods failed to insert. *

Furthermore, per FSAR Section 3.5.3, the worth of the RSD system with the maximum worth RSD unit inoperable is at least 0.12 delta k in the initial core and 0.11 delta k in the equilibrium core. This is sufficient to ensure shutdown during the first 2 weeks 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.

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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 P/F RATIO experienced during the transient (or the transient may be divided into smaller P/F RATIO intervals), as follows: *

   a) P/F RATIOS Less Than or Equal to 2.5 and above the limit of Figure 3.2.6-1: *

      For each P/F RATIO interval experienced during the transient, the fraction of Allowable Operating Time shall be the transient time that the P/F RATIO is within the bounds of the P/F RATIO interval, divided by the Allowable Operating Time per Figure 3.2.6-2 (for each interval) based on the maximum P/F RATIO experienced during the interval. *

   b) P/F 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 P/F 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 P/F RATIOS less than 2.5 are given in c.1.a above.

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c) P/F 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 P/F 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 P/F 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 P/F 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 P/F 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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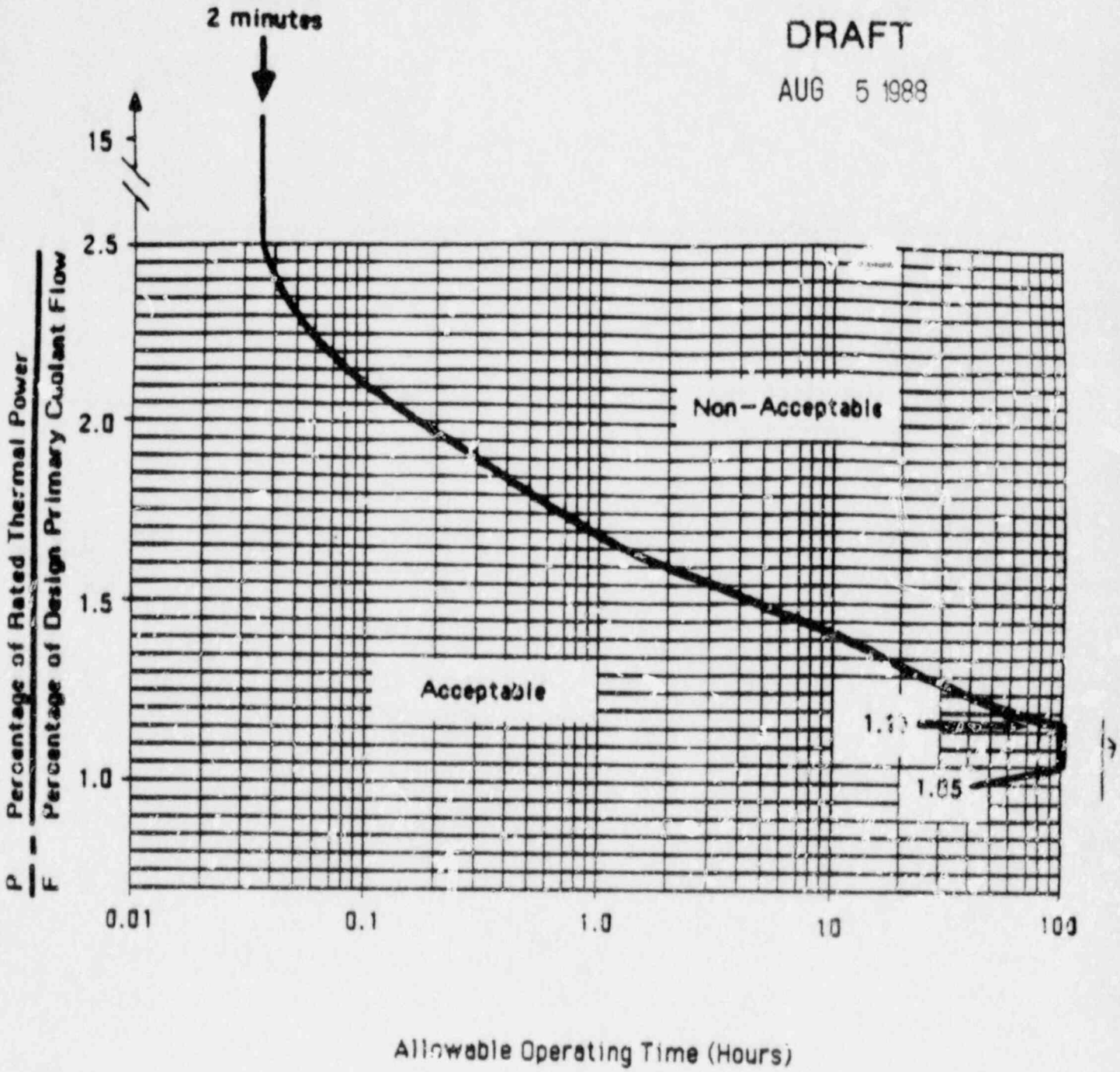

ALLOWABLE OPERATING TIMES WITH
POWER - TO - FLOW RATIOS EXCEEDING FIGURE 3.2.6-1

Figure 3.2.6-2

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BASIS FOR SPECIFICATION LCO 3.2.6/SR 4.2.6 (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 for 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 any P/F RATIO transient, the actual transient time period for each P/F RATIO interval during which the limit of Figure 3.2.6-1 is exceeded, 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-FLOW RATIOS less than or equal to 2.0.

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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 2.2.1-1-------	*
7.	Primary Coolant Pressure -Programmed High	---------See Table 2.2.1-1-------	*
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

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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.

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)

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.

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BASIS FOR SPECIFICATION LCO 3.3.1/SR 4.3.1 (Continued)

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 alternately, 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 alternately, 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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BASIS FOR SPECIFICATION LCO 3.3.2.2/SR 4.3.2.2 (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 Specification 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.

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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. *

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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 Sr-90 bone dose equivalent total plateout 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.

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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 bone dose equivalent 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.a.

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.

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BASIS FOR SPECIFICATION LCO 3.4.2/SR 4.4.2 AUG 5 1988

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 Sr-90 bone dose equivalent 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:

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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/cycle. 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 cycle 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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PLANT AND SAFE SHUTDOWN COOLING SUPPORT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.7.1.5 a. At least one steam generator economizer-evaporator-superheater (EES) safety valve per operating loop (V-2214, V-2215, V-2216, V-2245, V-2246, or V-2247) shall be OPERABLE for each boiler feed pump in operation supplying feedwater to the EES sections. OPERABLE valve setpoints shall be in accordance with Table 4.7.1-1.* *

b. Both reheater safety valves (V-2225 and V-2262) shall be OPERABLE with setpoints in accordance with Table 4.7.1-1.*

APPLICABILITY: POWER, LOW POWER and STARTUP

ACTION: a. With one or more of the above required EES safety valves inoperable in any one loop or with one reheater safety valve inoperable, restore the required valve(s) to OPERABLE status within 72 hours or restrict plant operation as follows: *

1. With an EES safety valve(s) inoperable, restrict plant operation so that the number of boiler feed pumps in operation corresponds to the number of OPERABLE safety valves as required above.

2. With a reheater safety valve inoperable, be in at least SHUTDOWN within the next 24 hours.

b. With one or more of the above required EES safety valves inoperable in both loops, restore the required valve(s) to OPERABLE status within 12 hours or restrict plant operation so that the number of boiler feed pumps in operation corresponds to the number of OPERABLE safety valves as required above. *

c. The provisions of Specification 3.0.4 are not applicable.

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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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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.

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

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 with only one boiler feed pump supplying feedwater to the EES sections. Each boiler feed pump is capable of supplying approximately one-third of the full power feedwater requirements (FSAR Section 10.2.3.1). As additional boiler feed pumps are placed in service, additional safety valves are also placed in service. The use of one safety valve per steam generator section during SHUTDOWN and REFUELING is acceptable, as it is capable of relieving the available flow. Also, other power actuated valves that are capable of relieving pressure from the main steam and reheat piping are included in the FSV design.

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.

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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 the single largest load 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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BASIS FOR SPECIFICATION LCO 3.8.1/4.8.1 (Continued) AUG 5 1988

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". In SR 4.8.1.1.2.e.3, the single largest load to be rejected is a circulating water pump, which is rated at 202 KW. *

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

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TABLE 4.8.4-1

ACM DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests	Test Frequency	
≤ 1	At least once per 31 days	*
≥ 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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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 conducted from the Refueling floor *

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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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.

*

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 cycle, in conjunction with the remaining fuel in the core, 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 an 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.

d. Ensure temperature coefficients at least as negative as those used in the FSAR analysis, throughout the cycle. *

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