UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

PUBLIC SERVICE COMPANY OF COLORADO

DOCKET NO. 50-267

FORT ST. VRAIN NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
License No. DPR-34

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Public Service Company of Colorado (the licensee) dated September 11, 1975; December 1, 1975; March 29, 1976; and June 14, 1976, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Special Reactors Branch
Division of Project Management

Attachment:
Change to the Technical
Specifications

Date of Issuance: JUN 1 8 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 13

FACILITY OPERATING LICENSE NO. DPR-34

DOCKET NO. 50-267

Replace the following existing pages with the attached revised pages bearing the same numbers:

4.1-13 thru 4.1-15
4.2-13 thru 4.2-15a
4.4-3, 4.4-8, 4.4-8a, and 4.4-15
4.9-3
5.1-4 thru 5.1-6
5.4-4 and 5.4-16
7.6-10

The changes are noted by marginal lines.

Add new pages 4.1-16, 4.2-15b, 4.3-6, 4.4-16, and 4.9-4.

JUN 1 8 1976

more than two feet into the core, is no greater than 50°F above the average core outlet temperature and which, for the remaining regions, is no greater than 200°F above the core average outlet temperature. A measurement uncertainty of ± 50°F was assumed for the core region outlet temperatures in the development of Specification SL 3.1. Specifying these maximum deviations from the average core outlet temperature will assure that the criteria upon which Specification SL 3.1 is based is met.

During power operation with an average core outlet temperature less than 950°F, sufficient overcooling of the core is provided with a +400°F deviation between the maximum and average core outlet temperature to assure that Specification SL 3.1 remains valid and that the integrity of the fuel particles would be preserved.

The time at temperatures exceeding the limits given represents conditions significantly below the core safety limit.

Specification LCO 4.1.8 - Reactivity Status, Limiting Conditions for Operation

If the difference between the observed and the expected reactivity, based on normalization to a base steady core condition, reaches 0.01 Δk, the reactor shall be shut down and reactor operations shall not be resumed until permission is received from the NFSC.

The initial base steady state core condition and changes of this base shall be approved by the NFSC.

Changes to the base approved by the NFSC shall be reported immediately to the Director, Office of Nuclear Reactor Regulation.

Basis for Specification LCO 4.1.8

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 Δk would be unexpected, and its occurrence would be thoroughly investigated and evaluated. The value of 0.01 Δk is considered to be a safe limit since a shutdown margin of at least 0.01 Δk with the highest worth rod pair fully withdrawn is always maintained (see LCO 4.1.2).

Specification LCO 4.1.9 - Core Region Temperature Rise, Limiting Condition for Operation

Whenever the reactor is pressurized to more than 50 psia and the core inlet orifice valves are set for equal region coolant flows, the reactor helium coolant flow shall be above the minimums given in Figure 4.1.9-1 (at the appropriate power level). Whenever the reactor is pressurized to more than 50 psia and the core inlet orifice valves are set at any positions other than for equal region flows, the measured helium coolant temperature rise through any core region shall not exceed the limits given in Figure 4.1.9-2 (at the appropriate power level). Below 50 psia reactor pressure, the maximum measured region helium coolant temperature rise shall not exceed 350°F with the core inlet orifice valves set at any position, and shall not exceed 600°F with the core inlet orifice valves set for equal flow.

If the measured helium coolant temperature rise exceeds these limits, immediate corrective action shall be taken. If this corrective action is not successful within fifteen (15) minutes, an immediate orderly shutdown shall be initiated.

Basis for Specification LCO 4.1.9

A maximum core region helium coolant temperature rise as a function of calculated reactor thermal power (including power from decay heat) has 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 potentials for possible local helium flow stagnation, which could result in excessive fuel temperatures.

The maximum core region helium temperature rise limits have been developed based upon a number of conservative assumptions. It was assumed that the primary system was pressurized to full inventory. At less than full inventory, higher region delta T's are acceptable. The core inlet helium temperature was assumed to be 100°F. At higher core inlet temperatures, higher region delta T's are acceptable. For the condition with helium flow orifice valves adjusted to yield equal helium flow to all fuel elements, a 25% margin of safety was added to the minimum flow requirements. For the condition with orifice valves at any position, the allowable region delta T is based upon a region power density (P_{reg}/P_{core}) equal to 0.4. For regions with higher power densities, higher region delta T's are acceptable. Conservatively these have been restricted to those of an 0.4 power density region. For regions with power densities between 0.4 and 0.28, when the flow control valves are fully closed, the region delta T's would not exceed their corresponding limits as a result of bypass flow around the closed orifice valve.*

---

*GA-9720, "PSC Quarterly Progress Report for the period ending September 30, 1969," October 29, 1969.

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

Specification LCO 4.2.10 - Loop Impurity Levels, High Temperatures, Limiting Conditions for Operation

The reactor shall not be operated with an average core outlet temperature $\geq$ 1200°F, if chemical impurity concentrations in the primary coolant exceed 10 ppm (by volume) for the sum of H_2O, CO, and CO_2. However, these amounts may be exceeded by up to a factor of 10 for a period of ten days, or by up to a factor of 100 for one day from the time the limit is exceeded.

Basis for Specification LCO 4.2.10

For plant operation in the normal power range (25% to 100% of rated thermal power), maximum impurity levels have been established to restrict carbon transport from the reactor core to cooler portions of the primary coolant system to about 330 lb/yr.

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

Specification LCO 4.2.11 - Loop Impurity Levels, Low Temperatures, Limiting Conditions for Operation

With the reactor operating and an average core outlet temperature below 1200°F, impurity levels shall not be allowed to exceed:

H_2O - dew point limits as a function of average core outlet temperature given in Figure 4.2.11-1.

CO_2 - 1000 ppm (by volume)

CO - 15,000 ppm (by volume)

In addition to those limits above, during reactor startups and shutdowns, the total time when reactor average outlet temperatures are between 725°F and 1200°F, and the moisture dew point is higher than -20°F shall not exceed a total of 90 days during any one refueling cycle.

Basis for Specification LCO 4.2.11

During plant startups, core average outlet temperatures will be below 1200°F until the final stages when steam temperatures are increased to rated and the plant enters the normal power range. At these lower temperatures, graphite corrosion by the various chemical impurities is minimal and there is reduced concern for carbon transport.

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

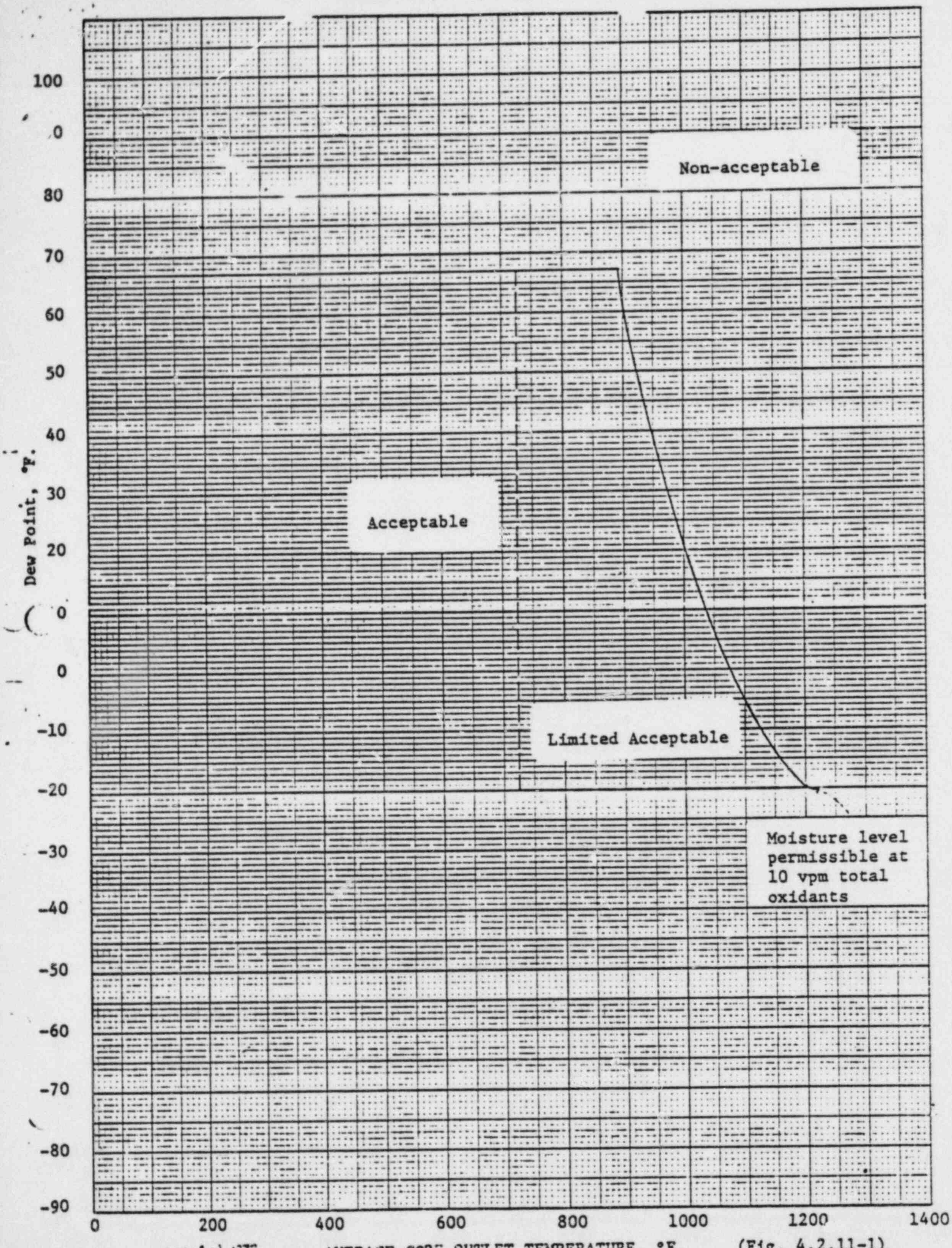
In the presence of moisture, boron carbide B_4C, is subject to oxidation at a temperature-dependent rate to form boron oxide, B_2O_3. In the event of subsequent significant steam inleakage the boron oxide is converted to volatile boric acid, which is capable of being steam-distilled from the core. Such 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°F that the rate of oxidation of boron carbide becomes sufficient to become a limiting parameter. At average core outlet temperatures above 1200°F, however, boron oxidation is of reduced significance because:

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

2) since the reactor must be at power to develop sustained temperatures in this range, the rate of B^{10} depletion by burnup significantly exceeds the rate of B_4C oxidation for allowable impurity levels (LCO 4.2.10).

The criterion used to establish the limits of Figure 4.2.11-1 in the range from 725°F to 1200°F was that not more than 10% of the beginning of life (BOL) B can be present as oxide over a refueling cycle. This criterion is based on the BOL B worth of 0.06 Δk, and the fact that 10% worth, 0.006 Δk, is substantially less than the minimum core shutdown margin of 0.016 Δk (FSAR Section 3.5.3.1), and only one-half of the reactivity anomaly of 0.012 Δk specified in Technical Specification SR 5.4.1.

The stipulation used in developing the curve of Figure 4.2.11 is that with the reactor outlet temperature in the range between 725°F and 1200°F,and with a primary coolant dew point temperature higher than -20°F, operating time under these conditions would be limited to no more than 90 days over any refueling cycle. By combining this stipulation with the dew point limits shown in Figure 4.2.11-1 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.


Non-acceptable
Acceptable
Limited Acceptable
Moisture level permissible at 10 vpm total oxidants
Dew Point, °F.
100
90
80
70
60
50
40
30
20
10
0
-10
-20
-30
-40
-50
-60
-70
-80
-90
0
200
400
600
800
1000
1200
1400
AVERAGE CORE OUTLET TEMPERATURE, °F.

(Fig. 4.2.11-1)

The dew point limit of 67°F below 725°F average outlet tempera- ture 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°F.

Specification LCO 4.2.12 - Liquid Nitrogen Storage, Limiting Conditions for Operation

The reactor shall not be operated at power if the liquid nitrogen storage tank level drops below 500 gallons.

Basis for Specification LCO 4.2.12

Adequate liquid nitrogen storage is provided to permit depressuri- zation of the PCRV via the helium purification system, assuming complete loss of all nitrogen recondensing capability. (FSAR, Section 9.6.6).

Continued cooling of the low temperature adsorbers is not required in the event all refrigeration is lost, insofar as the heatup due to decay heat would take more than a week to reach a temperature level above design conditions. This source of heat can be used to regenerate the LTA, transfixing the source of heat to the waste system. (FSAR, Section 9.4.6).

Specification LCO 4.2.13 - PCRV Liner Cooling System, Limiting Conditions for Operations

At least one heat exchanger and one pump shall be operating in each of the two PCRV liner cooling water loops during power operation. If this condition cannot be met during power operation, the following action shall be taken:

Specification LCO 4.3.9 - High Pressure Helium Supply System

During power operation, the high pressure helium supply shall be manually isolated from the helium circulator buffer supply header via valves V-23221 and V-23224.

Basis for Specification LCO 4.3.9

The need for isolation of the high pressure supply from the helium circulator buffer supply header arises from the possibility of upsetting the auxiliary system of one or more circulators by the unwanted introduction of high pressure gas. Isolating the high pressure supply therefore reduces the possibility of upsetting the buffer seal on helium circulators in both loops.

Specification LCO 4.4-1

TABLE 4.4-1

INSTRUMENT OPERATING REQUIREMENTS FOR PLANT PR 'ECTIVE SYSTEM, SCRAM

NO.	FUNCTIONAL UNIT	TRIP SETTING	MINIMUM OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS
1a.	Manual (Control Room)	—	1	0	None
1b	Manual (Emergency Board)	—	2 (f)	1	None
2.	Startup Channel-High	≤ 10^5 cps	2	1	Reactor Mode Sw. in "RUN"
3a.	Linear Channel-High, Channels 3, 4, 5	≤ 140% power (a)	2 (f)	1	None
3b.	Linear Channel-High, Channels 6, 7, 8	≤ 140% power (a)	2 (f)	1	None
4.	Primary Coolant Moisture High Level Monitor Loop Monitor	 ≤67°F Dewpoint ≤27°F Dewpoint	 1 (f, t) 2/Loop (f, t)	 1 (c) 1/Loop	 None (h)
5.	Reheat Steam Temperature - High (b)	≤ 1075°F (a)	2 (b) (f)	1	None
6.	Primary Coolant Pressure - Low	≤ 50 psig below normal, load programmed (a)	2 (f) (k)	1	Less than 30% rated power
7.	Primary Coolant Pressure - High	≤ 7.5% above normal rated, load programmed (a)	2 (f) (k)	1	None
8.	Hot Reheat Header Pressure - Low	≥ 35 psig	2 (f)	1	Less than 30% rated power
9.	Main Steam Pressure - Low	≥ 1500 psig	2 (f)	1	Less than 30% rated power
10.	Plant Electrical System-Loss	(d)	2 (e) (f)	1	None
11.	Two Loop Trouble	—	2	1	Reactor mode switch in "Fuel Loading"
12.	High Reactor Building Temperature (Pipe Cavity)	≤ 325°F	2 (f)	1	None

JUN 1 8 1975

Specification LCO 4.4.1

NOTES FOR TABLES 4.4-1 THROUGH 4.4-4

(a) See Specification LSSS3.3 for trip setting.

(b) 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.

(c) 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.

(d) Both 480 volt buses 1A and 1C less than 60% normal voltage for longer than 30 seconds.

(e) One channel consists of one undervoltage relay from each of the two 480 volt buses (two undervoltage relays per channel). These relays fail open which is the direction required to initiate a scram.

(f) The inoperable channel must be in the tripped condition, unless the trip of the channel will cause the protective action to occur.

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

(h) Permissible Bypass Conditions:

I. Any circulator buffer seal malfunction.
II. Loop hot reheat header high activity.
III. As stated in LCO 4.9.2.

(j) Items 1a. or 1c. or 1d. accompanied by 2a., 2b., 2c., or 2d. on Table 4.4-2 are required for loop 1 shutdown. Items 1b. or 1c. or 1f., accompanied by 2a., 2b., 2c., or 2d. on Table 4.4-2 are required for loop 2 shutdown.

(k) One operable helium circulator inlet thermocouple in an operable loop is required for the channel to be considered operable.

(m) Low Power RWP bistable resets at 4% after reactor power initially exceeds 5%.

(n) Power range RWP bistable resets at 10% after reactor power initially exceeds 30%.

(p) Item 7a. must be accompanied by item 7c for Loop 1 shutdown. Item 7b. must be accompanied by item 7c. for loop 2 shutdown.

Notes for Tables 4.4-1 through 4.4-4 (continued)

(r) Separate instrumentation is provided on each circulator for this functional unit. Only the affected helium circulator shall be shut down within 12 hours if the indicated requirements are not met.

(s) Each channel has 2 microphones running in parallel with one ultrasonic amplifier. For the channel to be considered operable, both microphones and the amplifier must be operable.

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

1)

Reactor Power Range	Minimum Sample Flow
Startup to 2%	1 scc/sec.
>2% - 5%	5 scc/sec.
>5% - 20%	15 scc/sec.
>20% - 35%	30 scc/sec.
>35% - 50%	50 scc/sec.
>50% - 100%	To be determined prior to exceeding 50% power

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°F.

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

Specification LCO 4.4.4 - Seismic Instrumentation - Limiting Conditions for Operation

The reactor shall not be operated at power unless three (3) of the six (6) seismic instruments are operable.

Basis for Specification LCO 4.4.4

The monitoring provided by three (3) seismic instruments, in the event of an earthquake, is adequate to determine the ground acceleration at the site.

Specification LCO 4.4.5 - Analytical System Primary Coolant Moisture Instrumentation - Limiting Condition for Operation

The reactor shall not be operated between a shutdown condition and 5% power during startup unless the primary coolant is being sampled for moisture by two analytical system moisture monitors.

If one monitor should become inoperable while increasing reactor power between shutdown and 5%, the second monitor shall be made operable or the reactor shall be shut down within 12 hours.

If both of the monitors become inoperable, during the above mentioned power increase, the reactor shall be shut down immediately.

During reactor power reduction from 5% power to shutdown conditions, at least one analytical system moisture monitor must be in operation. If both monitors become inoperable, the reactor shall be shut down immediately.

Basis for Specification LCO 4.4.5

During reactor operation, the analytical system primary coolant moisture monitor system is required below 5% reactor power for administration of LCO 4.2.11. One monitor is sufficient to detect primary coolant moisture content on a continual basis.

However, two monitors will normally be in service sampling primary coolant. These monitors do not provide any automatic action (other than an alarm function). Operator action is required to take corrective action in the event of high moisture levels in the primary coolant in the shutdown to 5% reactor power range.

Operator reaction time 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 are acceptable. As indicated in Document GA-A13677, Test and Evaluation of the Fort St. Vrain Dew Point Moisture Monitors System, one of the limiting parameters for determining required response times to shut the reactor down in the event of high primary coolant moisture is graphite oxidation. The allowable weight loss of the hottest fuel element in the core is 1%.

At operating temperatures experienced at 5% reactor power, response times to scram the reactor to limit oxidation to 1% by weight is approximately 6700 seconds, well within the capabilities of an operator.

Following completion of Phase 1 activities, and the appropriate approval by the AEC's Regional Regulatory Operations Office, Specification LCO 4.9.1 shall not restrict operation of this facility.

Basis for Specification LCO 4.9.1

The following of the two-phase approach identified ensures that all system modifications, testing, and documentation necessary to protect the health and safety of the public are completed in an orderly and timely manner.

Specification LCO 4.9.2 - Plant Protection System Dew Point Moisture Monitor Tests During Phase 2

During the "B" series of startup tests, the Plant Protective System (PPS) moisture monitors shall be tested at power levels of 5%, 25%, and 100% by injection of moisture-laden gas into the primary coolant. Moisture injection tests may be made at intermediate power levels if additional data is needed. These tests shall be conducted with the moisture monitor input trip functions to the Plant Protective System which cause scram, loop shutdown, circulator trip, and steam water dump, disabled. In addition, the Analytical System moisture monitors shall be utilized to monitor primary coolant moisture during these tests.

During the time that the Plant Protective System moisture monitor trips are disabled, two observers in direct communication with the reactor operator shall be positioned at the moisture monitor location on control board I-9310. These observers shall continously monitor the primary coolant moisture level and alert the reactor operator to any indicated moisture change.

Basis for Specification LCO 4.9.2

Public Service Company of Colorado has agreed to perform response time tests on the Plant Protective System dew point moisture monitor system during the rise-to-power program for Fort St. Vrain, Unit No. 1. These tests are intended to determine the moisture monitoring system response times under actual primary coolant system operating conditions.

The tests will include injection of moisture-laden helium into the sample rake of at least one helium circulator in each loop to determine system response times. Moisture injections are to be made at 5%, 25%, and 100% of rated reactor thermal power.

To facilitate this testing, it will be necessary to temporarily remove the dew point moisture monitors from service, thereby removing their inputs from the Plant Protective System. Removal of these inputs to the Plant Protective System will prevent automatic scram, loop shutdown, circulator trip, or steam/water dump due to high primary coolant system moisture.

Placing two observers at the moisture monitor location in the control room will insure prompt corrective action will be taken if a moisture ingress situation occurs during the moisture injection test.

Special test precautions have been specified for monitoring primary coolant moisture content during these tests to assure compliance with LCO 4.2.10 and 4.2.11, as applicable.

Specification SR 5.1.3 - Temperature Coefficient Surveillance

The reactivity change as a function of core temperature change shall be measured at the beginning of each refueling cycle.

Basis for Specification SR 5.1.3

The major shifts in reactivity change as a function of core temperature change will occur following refueling. The specified frequency of measurement following each major refueling will assure that the change of reactivity as a function of changes in core temperature will be measured on a timely basis to evaluate the limit specified in Specification LCO 4.1.5.

Specification SR 5.1.4 - Reactivity Status Surveillance

A surveillance check of the reactivity status of the core shall be performed at each startup and once per week during power operation. If the difference between the observed and the expected reactivity, based on normalization to a base steady state core condition, reaches 0.01 Δk, this discrepancy shall be considered an abnormal occurrence.

The initial base steady state core condition and changes of this base shall be approved by the NFSC.

Basis for Specification SR 5.1.4

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

This specification is designed to ensure that the core reactivity level is monitored to reveal in a timely manner the existence of potential safety problems or operational problems. An unexpected and/or unexplained change in the observed core reactivity could be indicative of such problems.

The normalization to an initial base steady state core condition will eliminate discrepancies due to manufacturing tolerances, analytical modeling approximations and deficiencies in basic data at the beginning of operation. Changes of the base steady state core conditions are permissible to eliminate explainable discrepancies resulting from long-term reactivity burnup effects and core refuelings.

Comparison of predicted and observed reactivities in a base steady state configuration will ensure the comparison will be easily understood and readily evaluated.

Any reactivity anomaly greater than 0.01 Δk would be unexpected and its occurrence would be thoroughly investigated and evaluated. The value of 0.01 Δk is considered to be a safe limit since a shutdown margin of at least 0.01 Δk with the highest worth rod pair fully withdrawn is always maintained (see LCO 4.1.2).

Specification SR 5.1.5 - Withdrawn Rod Reactivity Surveillance

The reactivity worth of the control rods which are withdrawn from the low power condition to the operating condition, in the normal withdrawal sequence, shall be measured at the beginning of each refueling cycle. The measured rod worths will be used to insure that the criteria for the selection of the rod sequence of Specification LCO 4.1.3 are met.

Basis for Specification SR 5.1.5

The measurement of control rod worths at the beginning of a refueling cycle will provide for an evaluation of calculational methods for control rod worths used in the prediction of the maximum worth rod in Specification LCO 4.1.3.

Specification SR 5.1.6 - Core Safety Limit Surveillance

During power operation the total operating time of the fuel elements within the core at power-to-flow ratios above the curve of Figure 3.1-2 will be evaluated once per week when the plant operation is within the normal operating range, and as soon as practicable after any deviation from the normal operating range. These operating times will be compared to the allowable operating time of Specification SL 3.1 to assure that the Core Safety Limit has not been exceeded.

Basis for Specification SR 5.1.6

Only during operation of the plant outside of the normal operating range is there a potential for accumulating significant operating times at power-to-flow ratios greater than the curve of Figure 3.1-2. Therefore, weekly evaluations of the total accumulated operating time at power-to-flow ratios greater than the curve of Figure 3.1-2 is sufficient during normal operation. Following any significant deviation from the normal operating range, the operation should be evaluated to determine the degree to which the actual total operation of the core approached the Core Safety Limit.

Table 5.4-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTING OF SCRAM SYSTEM (Continued)

Channel Description	Function	Frequency (1)	Method
6. Continued	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 (t) of Notes for Tables 4.4-1 through 4.4-4.
	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.
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. Internal test signal to verify trips and alarms.
	c. Calibrate	R	c. Compare each thermocouple output with calibrated RTD. Internal 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, internal test signal to verify trips and alarms.
	c. Calibrate	R	c. Known pressure applied to sensor. Internal 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, internal test signal to verify trips and alarms.
	c. Calibrate	R	c. Compare thermocouple with calibrated RTD. Internal test signal to adjust trips and indicators.

Specification SR 5.4.11 - PCRV Surface Temperature Indication - Surveillance Requirement

The PCRV surface temperature indicators shall be functionally tested monthly and calibrated annually.

Basis for Specification SR 5.4.11

The PCRV surface temperature indicators provide for continuous monitoring of surface concrete temperatures to assure the proper temperature gradient is maintained through the PCRV wall and heads.

The surveillance interval specified is adequate to detect any drift or malfunction of this instrumentation.

Specification SR 5.4.12 - Analytical System Primary Coolant Moisture Instrumentation - Surveillance Requirements

The analytical system primary coolant moisture instrumentation shall be calibrated on a once per refueling cycle basis.

Basis for Specification SR 5.4.12

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

3. Post Irradiation Examination of Fuel Elements

Conduct examination, as soon as practical, of the burnable poison rods contained in irradiated fuel elements removed after the third refueling. This examination will determine the chemical composition of the boron remaining, and will be an indication of what effect the presence of any moisture in the primary coolant may have had on the boron content. Since the majority of the boron 10 isotope will have been depleted due to neutron irradiation, some uncertainty will be present in this analysis. The results of the post irradiation examination shall be reported as soon as it is available.

Basis for Specification AC 7.6

The information specified to be reported periodically by this Specification is adequate to document the operation of the plant related to safety.