UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

PUBLIC SERVICE COMPANY OF COLORADO

DOCKET 50-267

FORT ST. VRAIN NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License DPR-34

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

   A. The application for amendment by Public Service Company of Colorado (the licensee) dated October 15, 1982, as superseded by letter dated February 22, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's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. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License DPR-34 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 33, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

for G. L. Madsen, Chief
Reactor Project Branch 1

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 8, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE DPR-34

DOCKET 50-267

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
iv	iv
v	v
2-5	2-5
2-6	2-6
5-0.1	5-0.1
--	5-0.2
5.1-1 through 5.1-10	5.1-1 through 5.1-15
5.2-1 through 5.2-21(a)	5.2-1 through 5.2-43
5.3-1 through 5.3-8	5.3-1 through 5.3-17

PAGE

5.2 PRIMARY COOLANT SYSTEM - SURVEILLANCE REQUIREMENTS (Continued)

2.16 Refueling Shutdown

The reactor is considered shut down for refueling purposes when the reactor mode switch is locked in the "Fuel Loading" position simultaneous with either hot shutdown or the cold shutdown reactivity conditions.

2.17 Safe Shutdown Cooling

Safe shutdown cooling refers to cooling of the core with Safe Shutdown Equipment providing for removal of core stored energy and for adequate sustained decay heat removal. The reactivity condition in the core is either hot or cold shutdown.

2.18 Surveillance Interval

A surveillance interval is the interval of time between surveillance check, tests, or calibration. Unless otherwise stated, the surveillance interval can be adjusted by ± 25% to accommodate normal operational schedules. Unless otherwise stated in these specifications, surveillance may be terminated on those instruments or equipment not in normal use during reactor shutdown or refueling shutdown if the surveillance interval is one month or less.

2.19 Trip

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

2.20 Core Average Outlet Temperature

The core average outlet temperature is defined as the arithmetic average of the individual refueling region outlet temperatures,

$$T\ avg = \frac{\sum_{i=1}^{37} Ti}{37}$$

where Ti = individual refueling region outlet temperature.

5.0 SURVEILLANCE REQUIREMENTS

The surveillance requirements specified in this section define the tests, calibrations, and inspections which are necessary to verify the performance and operability of equipment essential to safety during all modes of operation, or required to prevent or mitigate the consequences of abnormal situations.

Implementation of the in-service inspection (ISI) surveillance requirements shall be per one of the following criteria, unless otherwise indicated:

ISI Criterion A: The surveillance requirement shall be implemented before 90 days have elapsed following the formal approval date of Amendment No. 33 by the Nuclear Regulatory Commission.

ISI Criterion B: The surveillance requirement shall be implemented before the beginning of fuel cycle 4, provided that fuel cycle 4 does not begin within 90 days from the formal approval date of Amendment No. 33 by the Nuclear Regulatory Commission.

Otherwise, the surveillance requirement shall be implemented before the end of the first scheduled plant shutdown following 90 days from the formal approval date of

Amendment No. 33 by the Nuclear Regulatory Commission.

ISI Criterion C: The surveillance requirement shall be implemented before the beginning of fuel cycle 5.

ISI Criterion D: The surveillance requirement shall be implemented in the existing schedule of surveillance tests, following 90 days from the formal approval date of Amendment No. 33 by the Nuclear Regulatory Commission.

5.1 REACTOR CODE AND REACTIVITY CONTROL - SURVEILLANCE REQUIREMENTS

Applicability

Applies to the surveillance of the reactor core and core reactivity control mechanisms.

Objective

To ensure the capability to control the reactivity and temperature of the reactor core.

Specification SR 5.1.1 - Control Rod Drives Surveillance

The surveillance of the control rod drives shall be as follows:

a) All 37 control rod pairs will be scrammed from the full out to the full in position once a year and the scram time measured. Operable withdrawn control rods shall have a scram time less than 160 seconds.

b) All control rods which are withdrawn during power operation will be exercised a short distance (about 6 inches) once a month. Operation of position indicators, motion indicators, and the absence of slack cable indication shall be verified.

Basis for Specification SR 5.1.1

Tests will be performed on the control rod drives to assess their capability to control the reactivity of the reactor core. On a yearly basis, the control rods will be scrammed from the full out position and the scram time measured. The drive mechanisms are designed for a normal scram time of 140 $\pm$ 10 seconds. However, for safe reactivity control of the reactor, scram times of the drive mechanisms may be as great as 160 seconds without altering the kinetics of the scram.

The drive mechanism will be used to exercise sequentially, all withdrawn rods over a short distance (about 6 inches) once a month. This test will assess the operability of the control rods and drives and position indicating instrumentation. Any binding of the rods in their channels can be determined by a slack cable indication.

Specification SR 5.1.2 - Reserve Shutdown System Surveillance

The surveillance of the reserve shutdown system shall be as follows:

a) The ability to pressurize each of the 37 reserve shutdown hoppers to 10 psi above reactor pressure, as indicated by operation of the hopper pressure switch, shall be demonstrated every three months. Operable reserve shutdown hoppers shall be capable of being pressurized. The ability to operate the ACM quick disconnect valves, which provide an alternate means of actuating the hopper pressurization valves, shall be demonstrated every three months, and the ACM valve actuation gas pressure shall be monitored weekly.

SR 5.1.2.a shall be implemented per ISI Criterion A.

b) The test pressurizing gas pressure indicator shall be calibrated annually.

c) An off-line functional test of a reserve shutdown assembly shall be performed in the hot service facility, or other suitable facility, following each of the first five refueling cycles and at two refueling cycle intervals thereafter. These tests will consist of pressurizing reserve shutdown hopper to the point of rupturing the disc and releasing the poison material. If a reserve shutdown hopper rupture disc does not rupture at a differential pressure less than 300 psi and release the poison material, the reactor shall be placed in a shutdown condition until it can be shown that LCO 4.1.6 can be met.

d) The instrumentation which alarms a low pressure in the reserve shutdown actuating pressure lines shall be functionally tested in conjunction with the test, and at the same intervals, specified in part a) above, and calibrated once a year. Operable reserve shutdown hoppers shall have an actuating bottle pressure greater than or equal to 1,500 psig.

e) The reserve shutdown hopper pressure switches shall be calibrated at the same interval that they are removed from the reactor for maintenance.

f) Visual examination shall be performed of pipe sections which require disassembly and reassembly within the refueling penetrations, after they have been disassembled as required for refueling or maintenance.

SR 5.1.2.f shall be implemented per ISI Criterion B.

g) Demonstration shall be made at each refueling outage that each subsystem is operable by actuating each group of pressurizing valves from the Control Room. The capability of pressurizing the corresponding hoppers need not be demonstrated during this test. Valve position indication and fail safe operation shall be observed during this test.

SR 5.1.2.g shall be implemented per ISI Criterion B.

Basis for Specification SR 5.1.2

The reliability of the reserve shutdown system to perform its function will be maintained by a control system pressure test and actual off-line rupture tests conducted in the hot service facility or other suitable facility. The control system pressure test demonstrates the ability to pressurize the hoppers and indicates the operability of the control system components. A successful test will increase the hopper pressure about 10 psi above reactor pressure. This differential is well below the minimum 115 psi differential required to burst the disc.

The off-line tests consist of actual disc ruptures and poison drops. These will be used to determine the reliability of the differential burst pressure of the disc, and the tendency of the poison material to hang up or deteriorate in the hoppers over extended periods of time.

This test information will be used to verify the capability to shut down the reactor in an emergency situation. The reserve shutdown system hoppers operate in two subsystems. The first consists of the seven hoppers in refueling regions 1, 3, 5, 7, 22, 28, and 34; the second subsystem is comprised of the remaining 30 hoppers in the remaining refueling regions. Safe control of the reactor by the reserve shutdown system can be accomplished with one of the seven hoppers inoperative, and one of the remaining 30 hoppers inoperative. A differential pressure from 585 to 315 psi is available from the helium supply bottle with a pressure greater than or equal to 1,500 psig.

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

LCO 4.1.6 prevents performing an overall control system operational test at power since it allows only one reserve shutdown hopper to be inoperable in each subsystem when the reactor is either at low power or at power. To prevent the release of reserve shutdown material in the core, all hoppers of a subsystem must be rendered inoperable when testing the control system. This can only be performed when the reactor is shut down. Only valve actuation has to be tested since the ability to pressurize each hopper is demonstrated every three months.

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 $\Delta\kappa$ would be unexpected and its occurrence would be thoroughly investigated and evaluated. The value of 0.01 $\Delta\kappa$ is considered to be a safe limit since a shutdown margin of at least 0.01 $\Delta\kappa$ 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.

Specification SR 5.1.7 - Region Peaking Factor Surveillance

The calculated region peaking factors (RPF's) used in determining the individual region outlet temperatures for Regions 20 and 32 through 37 and percent RPF discrepancy (see LCO 4.1.7) for Regions 1 through 19 and 21 through 31 shall be evaluated according to the following schedule for each refueling cycle:

a) Calculated RPF's:

1) Prior to initial power operation after refueling.

2) At the equivalent of 20 ($\pm$5) effective days at

rated thermal power after refueling.

3) At the equivalent of 40 ($\pm$5) effective days at rated thermal power after refueling.

4) At monthly intervals thereafter, provided that the core has accumulated an exposure of at least the equivalent of 10 effective days at rated thermal power since the previous evaluation. If the core has accumulated an exposure of less than the equivalent of 10 effective days at rated thermal power since the previous evaluation, the evaluation may be deferred until the next applicable interval.

b) Percent RPF Discrepancy: Within a total elapsed time of 10 calendar days at reactor power levels above 40% of rated thermal power after the completion of any of the "Calculated RPF" evaluations required above with the following qualifications:

1) A "Percent RPF Discrepancy" evaluation shall be performed prior to exceeding 40% of rated thermal power for the first time after refueling, but at a reactor power above 30% of rated thermal power.

2) If the total elapsed time at reactor power levels above 40% of rated thermal power does not exceed 10 calendar days prior to the subsequent "Calculated RPF" evaluation, the

"Percent RPF Discrepancy" evaluation is not required, but the total elapsed time at reactor power levels above 40% of rated thermal power between "Percent RPF Discrepancy" evaluations shall not exceed 45 calendar days.

Basis for Specification SR 5.1.7

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

The calculated and measured region peaking factors for Regions 1 through 19 and 21 through 31 (candidate comparison regions) will change during the refueling cycle as fission product inventories saturate, fissile material and burnable poison are depleted, control rods are withdrawn from the core, and region flow characteristics change. A surveillance check of the percent region peaking factor discrepancy will provide assurance that the requirements of LCO 4.1.7c are being met for comparison regions. The frequency for surveillance has been established based upon conservative evaluations of potential fuel kernel migration, which could occur if a region with an excessively large, negative region peaking factor discrepancy were used as a comparison region.

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5.2 PRIMARY COOLANT SYSTEM - SURVEILLANCE REQUIREMENTS

Applicability

Applies to the surveillance of the primary (helium) reactor coolant system, excluding the steam generators.

Objective

To ensure the capability of the components of the primary reactor coolant system to maintain the primary reactor coolant envelope as a fission product barrier and to ensure the capability to cool the core under all modes of operation.

Specification SR 5.2.1 - PCRV and PCRV Penetration Overpressure Protection Surveillance

a) Each of the two overpressure protection assemblies protecting the PCRV shall be tested at intervals not to exceed five years, on an alternating basis, with one overpressure protection assembly tested during each refueling cycle.

The PCRV safety valve containment tank closure bolting shall be visually examined for absence of surface defects when the tank is opened for the above testing. Tank closure flange leak tightness shall be determined following tank closure.

SR 5.2.1.a shall be implemented per ISI Criterion C.

b) Each of the two overpressure protection assemblies protecting a steam generator or a circulator penetration interspace shall be tested at five calendar year intervals on an alternating basis, so that one safety valve for each penetration interspace and one rupture disc of each type are tested at an approximate interval of two and a half years.

SR 5.2.1.b shall be implemented per ISI Criterion D.

c) The instrumentation and controls associated with the overpressure protection assemblies in a) and b) above shall be tested and calibrated as follows:

1) The pressure switch and alarm for each interspace between a rupture disc and the corresponding safety valve shall be functionally tested monthly and calibrated annually.

The pressure switch and alarm for the PCRV safety valve containment tank shall be functionally tested and calibrated annually.

SR 5.2.1.c.1 shall be implemented per ISI Criterion D.

2) The position indication circuits associated with the PCRV overpressure protection system shut off valves shall be functionally tested and calibrated when testing either of the PCRV overpressure

protection assemblies. The pressure switch and alarm for the PCRV safety valve bellows shall be functionally tested and calibrated in conjunction with its associated safety valve test.

SR 5.2.1.c.2 shall be implemented per ISI Criterion C.

3) The control, interlock, and position indication circuits associated with each of the PCRV penetration overpressure protection system shut off valves shall be functionally tested at five calendar year intervals.

SR 5.2.1.c.3 shall be implemented per ISI Criterion D.

Basis for Specification SR 5.2.1

Testing of a PCRV overpressure protection assembly can only be performed when closing the corresponding manual shut off valve, located upstream of the rupture disc. LCO 4.2.7 does not allow isolation of such an assembly unless the primary pressure is less than 100 psia. Consequently, testing and examinations will be performed at shutdown. One assembly will be isolated while the other one will remain in a fully operational condition during the testing procedure, thus ensuring overpressure protection of the PCRV.

The rupture disc is designed to be removed from the system for bench testing. Verification is made of the correct deflection of the disc at the set pressure level which would cause the membrane to be ruptured. The safety valve is tested for setpoint activation without removing it from the system.

The pressurized portion of the assembly is monitored for leakage during plant operation. Leakage examination of the containment tank cover seals and visual examination of the cover bolts provides assurance that containment tank integrity is restored after the tank cover has been re-installed.

Testing of a PCRV penetration overpressure protection assembly can be performed during plant operation since the assemblies are accessible and since LCO 4.2.7 requires only one assembly to be operable at any time.

The safety valve in each assembly is tested while in place 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.

The intervals specified for testing the overpressure protection assemblies are adequate to demonstrate the operability of the overpressure protection systems.

The intervals specified for testing the associated instrumentation and controls are adequate to assure reliability of rupture disc and safety valve operation and to monitor the integrity of the PCRV safety valve piping and containment tank.

Specification SR 5.2.2 - Tendon Corrosion and Anchor Assemblies Surveillance

The serviceability of the corrosion protection applied to and the condition of the prestressing tendons shall be monitored in accordance with paragraphs a) and b). Surveillance of the tendon end anchor assemblies shall be performed in accordance with paragraph c).

a) Corrosion protected wire samples of sufficient length (i.e., initially at least 15 feet where practical, or half the tendon length, whichever is shorter) shall be inserted with selected tendons (those tendons with load cells). Corrosion inspection of at least one of these wires shall be made at the end of the first and third calendar year after prestressing. Additional inspections shall be conducted at five calendar year intervals thereafter.

   SR 5.2.2.a shall be implemented per ISI Criterion D.

b) A sample of the atmosphere contained in a representative number of tendon tubes (tendon tubes without load cells and tendon tubes with load cells from which wire samples are examined) shall be drawn and analyzed for products of corrosion, in coordination with and at the same time intervals as for paragraph a) above.

c) Visual examination of 5% of the prestressing anchor assemblies shall be performed at five calendar year intervals. This may include the anchor assemblies which can be visually examined while performing a) and b) above.

SR 5.2.2.c shall be implemented per ISI Criterion D.

<u>Basis for Specification SR 5.2.2</u>

The corrosion protection provided for the PCRV prestressing components is considered to be more than adequate to assure that the required prestressing forces are sustained throughout the operational life of the plant. The details of the corrosion protection system are described in Section 5.6.2.5 of the FSAR.

Sampling tendon tube atmosphere for products will provide a secondary check on the adequacy of the corrosion protection provided for the stressing tendons.

Visual examination of tendon end anchor assemblies will provide additional assurance that the prestressing system has not degraded by checking the corrosion protection and integrity of the anchor assemblies.

Specification SR 5.2.3 - Tendon Load Cell Surveillance

a) Checks on the possible shift in the load cell reference points for representative load cells shall be performed at the end of the first calendar year after initial prestressing and within 120 days prior to initial power operation. Additional checks shall be conducted at five calendar year intervals thereafter.

b) The load cell alarm circuit between the Data Acquisition System Room and the Control Room shall be functionally tested annually to assure that the operator in the Control Room is alerted when tendon load settings are exceeded.

SR 5.2.3.b shall be implemented per ISI Criterion A.

Basis for Specification SR 5.2.3

The PCRV tendons apply the force required to counteract the internal pressure. Therefore, they are the PCRV structural components most capable of being directly monitored and of indicating the capability of the vessel to resist internal pressures. Since the relation between effective prestress and internal pressure is directly and easily calculable, monitoring tendon loads is a direct and reliable means for assuring that the vessel always has capacity to resist pressures up to Reference Pressure.

Monitoring of the tendon loads will assure that deterioration of structural components including progressive tendon corrosion, concrete strength reduction, excessive steel relaxation, etc., cannot occur undetected to a degree that would jeopardize the safety of the vessel. Each of these phenomena would result in tendon load changes. These changes, as reflected by the load cells, are monitored in the control room by an alarm system which alerts the operator when the tendon load settings are exceeded. The upper settings will be varied depending on the location of the tendon being monitored, while the lower settings for all load cells will be set to correspond to 1.25 times peak working pressure (PWP).

Specification SR 5.2.4 - PCRV Concrete Structure Surveillance

a) Crack patterns on the visible surfaces of the PCRV shall be mapped prior to and following the initial proof test pressure (IPTP). Concrete cracks which exceed 0.015 inches in width shall be recorded. Subsequent concrete surface visual inspections shall be performed after the end of the first and third calendar year following initial power operation.

Recorded cracks shall be assessed for changes in length and any new cracks will be recorded. Additional inspections shall be conducted [illegible] ten calendar year intervals thereafter.

b) PCRV deformations and deflectio[illegible] midheight and at the center of the [illegible] be monitored at five calendar year [illegible] during a vessel pressurization to operating pressure.

SR 5.2.4.6 shall be implemented per ISI Criterion C.

c) The PCRV support structure shall be visually examined for evidence of structural deterioration at ten calendar year intervals.

SR 5.2.4.c shall be implemented per ISI Criterion C.

<u>Basis for Specification SR 5.2.4</u>

Cracks are expected to occur in the PCRV concrete resulting from shrinkage, thermal gradients, and local tensile strains due to mechanical loadings. The degree of cracking expected is limited to superficial effects and is not considered detrimental to the structural integrity of the PCRV. Reinforcing steel is provided to control crack growth development with respect to size and spacing. Model testing has also shown that severely cracked vessels contain the normal working pressure for extended periods of time as long as the effective prestressing forces are maintained.

Cracks up to about 0.015 inches (limits of paragraph 1508b, ACI 318-63) for concrete not exposed to weather are generally considered acceptable and corrosion of rebars at such cracks is of negligible consequence. Large crack widths will require further assessment as to their significance, depending on the width, depth, length, and location of the crack on the structure, and must be considered with reference to the observed overall PCRV response.

Further discussion on the significance of concrete cracks in the PCRV is given in Section 5.12.5 of the FSAR.

Observed crack development with time during reactor operation will be related to the PCRV structural response as monitored by the installed sensors and deflection measurements. Details of the PCRV structural monitoring provisions are given in Section 5.13.4 and Appendix E.17 of the FSAR.

The interval for surveillance after the fifth year following initial prestressing may be adjusted based on the analysis of prior results.

Monitoring of overall PCRV deformations and deflections is the best indication of PCRV structural performance and verifies that the PCRV response is elastic and that no significant permanent strains exist.

Visual examination of the PCRV support structure will indicate that no structural deterioration has occurred. Significant cracking patterns or sizes should be investigated with respect to their impact on the integrity of the PCRV.

Specification SR 5.2.5 - Liner Specimen Surveillance

Specimens shall be placed adjacent to the outside surface of the top head liner so that changes in notch toughness due to irradiation of the steel can be measured during the life of the reactor.

During the fifth refueling cycle, three sets of 12 specimens of the PCRV liner materials and weld material shall be removed and tested to obtain Charpy impact data. The specimen holders shall contain dosimeters to provide integrated neutron flux measurements. Additional specimen removal and testing shall be conducted during every tenth refueling cycle thereafter.

Basis for Specification SR 5.2.5

A test program will be performed to survey and assess the shifts in NDTT of the PCRV liner materials. The testing is to be accomplished by placing Charpy impact test specimens, made from the liner materials, near the liner and exposing them to appropriate neutron fluxes and temperatures. The Charpy impact test specimens are to be removed, 36 at a time, during the life of the vessel and tested to determine the condition of the vessel steel. The total number of specimens placed in the reactor is approximately 750, which will allow the determination of a complete impact transition curve for the plate metal, the weld metal and the heat affected zone at each test interval.

This testing program will meet the requirements of ASTM-E-185-70, with the following exceptions:

a) Tensile specimens are not included, since the liner is not a load carrying member, but only a ductile membrane.

b) No thermal control specimens have been provided, since there is no appreciable temperature cycling of the liner. The liner materials will normally be kept at or below 150 degrees fahrenheit during all plant operations.

Tests performed on this liner material (see FSAR Section 5.7.2.2) have indicated that no observable changes in material characteristics developed during an exposure to a fluence equivalent to the first five years of full power operation. Further, these tests demonstrated no significant damage after a fluence equivalent to 30 years of full power operation. The testing program prescribed for the Fort St. Vrain liner is in compliance with the ASME Boiler and Pressure Vessel Code, Section III N-110.

The interval for specimen removal and testing subsequent to the fifth refueling cycle may be adjusted based on the analysis of prior results.

Specification SR 5.2.6 - Plateout Probe Surveillance

One plateout probe shall be removed for evaluation coincident with the first, third, and fifth refueling, and at intervals not to exceed five refueling cycles thereafter. If, during the second or fourth refueling cycle, or any refueling cycle following the fifth refueling, the primary coolant noble gas activity (gamma + beta) should increase by 25% over the average activity of the previous three months at the same reactor power level and the primary coolant activity is greater than 25% of design, the plateout probe shall be removed at the end of that refueling cycle. The probes shall be analyzed for ^{90}Sr inventory in the reactor circuit. The probes removed shall also be analyzed for ^{131}I.

Basis for Specification SR 5.2.6

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

The probes shall be analyzed for ^{90}Sr and the results shall be used to establish the total ^{90}Sr inventory in the reactor circuit to determine compliance with LCO 4.2.8. Results of probe analyses shall be compared with the calculated estimates of ^{90}Sr which were made between probe removals. The analysis for ^{131}I shall be made to determine the degree of conservatism of the assumptions made regarding the circulating and plated out iodine in the primary coolant circuit.

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

Specification SR 5.2.7 - Water Turbine Drive Surveillance

Components of the helium circulator water turbine drive system shall be tested as follows:

a) One circulator and the associated water supply valving in each loop will be functionally tested by operation on water turbine drive using feedwater, condensate, and boosted condensate (supplied to the firewater booster pumps at fire pump discharge pressure), annually.

b) Safety valves (V-21522, V-21523, V-21542, and V-21543), located in the water turbine supply lines, will be tested for relieving pressure annually.

c) Both turbine water removal pumps and the turbine water removal tank overflow to the reactor building sump shall be functionally tested every three months.

d) The instrumentation and controls associated with c) shall be functionally tested in conjunction with and at the same intervals as the turbine water removal pumps and shall be calibrated annually.

Basis for Specification SR 5.2.7

The circulator water turbine drives are normally operated during an extended shutdown. Therefore the specified surveillance requirements are adequate to ensure water turbine operability.

Specification SR 5.2.8 - Bearing Water Makeup Pump Surveillance

The circulator bearing water makeup pumps and associated instruments and controls shall be tested as follows:

a) Normal Makeup Pump shall be operated in the recycle mode every three months.

b) Emergency Makeup Pump shall be functionally tested every three months.

c) The associated instruments and controls shall be functionally tested in conjunction with and at the intervals specified in parts a) and b) above, and calibrated annually.

Basis for Specification SR 5.2.8

During accident conditions described in FSAR Section 10.3.9, the circulator bearing water makeup pump is required to operate intermittently to make up bearing water. The specified testing interval is sufficient to ensure proper operation of the pumps and associated controls.

Specification SR 5.2.9 - Helium Circulator Bearing Water Accumulators Surveillance

The helium circulator bearing water accumulators, instrumentation, and controls shall be functionally tested monthly and calibrated annually.

Basis for Specification SR 5.2.9

Helium Circulator bearing water is normally supplied from the bearing water system and is backed up by the backup bearing water system supplied from the Emergency Feedwater Header. In the event of a failure in both of these systems, the water stored in the bearing water accumulators is adequate to safety shut down both helium circulators in a loop. The monthly test interval and annual calibration interval will assure proper operation of the accumulator controls if they should ever be called upon to function.

Specification SR 5.2.10 - Fire Water System/Fire Suppression Water System Surveillance

a) The fire water system shall be verified operable as follows:

1) The motor driven and engine driven fire pumps shall be functionally tested monthly. The associated instruments and controls shall be functionally tested monthly and calibrated annually.

2) The diesel engine fuel shall be inventoried monthly and sampled and tested quarterly.

3) The diesel engine shall be inspected during each refueling shutdown.

4) The diesel engine starting battery and charger shall be inspected weekly for proper electrolyte level and overall battery voltage. The battery electrolyte shall be tested quarterly for proper specific gravity.

5) The batteries, cell plates, and battery racks, shall be inspected each refueling cycle for evidence of physical damage or abnormal degradation. The battery-to-battery and terminal connections shall be verified to be clean, tight,

free of corrosion, and coated with anti-corrosion material each refueling cycle.

b) The fire suppression water system shall be verified operable as follows:

1) Monthly by verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

2) Semi-annually by performance of a fire suppression water system flush.

3) Annually by cycling each testable valve in the fire suppression water system flow path through at least one complete cycle of full travel.

4) Each refueling cycle by performing a fire suppression water system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

(a) Verifying that each automatic valve in the flow path actuates to its correct position.

(b) Verifying that each fire water pump develops at least 1,500 gpm at a system head of 290 feet.

(c) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

(d) Verifying that each fire water pump starts sequentially to maintain the fire suppression water system pressure at greater than or equal to 125 psig.

5) Each three years by performing a flow test.

<u>Basis for Specification SR 5.2.10</u>

The fire water pumps are required to supply water for fire suppression and safe shutdown cooling. The specified testing interval is sufficient to ensure proper operation of the pumps and controls. The motor driven pump routinely operates intermittently.

The operability of the fire suppression water system ensures that adequate fire suppression and emergency safe shutdown cooling capability is available. The specified testing interval is sufficient to ensure proper operation of the system when required.

Specification SR 5.2.11 - Primary Reactor Coolant Radioactivity Surveillance

A grab sample of primary coolant shall be analyzed a minimum of once per week during reactor operation for its radioactive constituents and shall be used to calibrate the continuous primary coolant activity monitor.

If the continuous primary coolant activity monitor is inoperable, the primary coolant activity level reaches 25% of the limits of LCO 4.2.8, or the primary coolant activity level increases by a factor of 25% over the previous equilibrium value of the same reactor power level, the frequency of sampling and analysis shall be increased to a minimum of once each day until the activity level decreases or reaches a new equilibrium value (defined by four consecutive daily analyses whose results are within ± 10%) at which time weekly sampling may be resumed.

Basis for Specification SR 5.2.11

The design of the instrumentation is such that under normal operating conditions the activity of the primary coolant is measured and indicated on a continuous basis. The weekly sampling interval provides an adequate check on the continuous monitoring equipment.

Specification SR 5.2.12 - Primary Reactor Coolant Chemical Surveillance

The primary coolant shall be analyzed for chemical constituents a minimum of once per week. If the chemical impurity levels exceed 50 percent of the limits of LCO 4.2.10 or LCO 4.2.11, whichever is applicable, the frequency of sampling and analysis shall be increased to a minimum of once each day until the level decreases or reaches a new equilibrium value (defined by four consecutive daily analyses whose results are within $\pm$ 10%), at which time weekly sampling may be resumed.

Basis for Specification SR 5.2.12

The chemical constituents in the primary coolant are routinely measured on a continuous basis. The specification of an interval for surveillance allows for routine maintenance of the chemical impurity monitoring equipment. The presence of higher than nominal impurity levels of chemical impurities is related to core materials corrosion which might occur only with very high levels for sustained periods of time.

Specification SR 5.2.13 - PCRV Concrete Helium Permeability Surveillance

The permeability of the PCRV concrete to helium shall be measured prior to the initial startup of the reactor and after the end of the third year following initial power operation. Additional measurements shall be made at five year intervals thereafter.

Basis for Specification SR 5.2.13

Measurements of the relative helium permeability throughout plant life provides, as a supplement to other surveillance efforts, information concerning the continued integrity of the PCRV concrete.

The interval for surveillance after the fifth year following the initial power operation may be adjusted based on the analysis of prior results.

Specification SR 5.2.14 - PCRV Liner Corrosion Surveillance Requirement

The PCRV liner shall be examined for corrosion induced thinning, using ultrasonic inspection techniques at the end of the third and fifth years following initial power operation. Additional examinations shall be conducted at ten year intervals thereafter.

Basis of Specification SR 5.2.14

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 provisions are discussed in Section 5.13 of the FSAR.

The interval for surveillance after the fifth year following initial power operation may be adjusted based on the analysis of prior results.

Specification SR 5.2.15 - PCRV Penetration Interspace Pressure Surveillance

The instrumentation which monitors the pressure differential between the purified helium supply header to the PCRV penetration interspaces and the primary coolant system will be functionally tested once every month and calibrated annually.

Basis for Specification SR 5.2.15

This calibration and test frequency is adequate to insure that the purified helium being supplied to the PCRV penetration interspaces shall be at a higher pressure than the primary coolant pressure within the PCRV.

Specification SR 5.2.16 - PCRV Closure Leakage, Surveillance Requirements

The surveillance of PCRV closure leakage shall be as follows:

a) PCRV primary and secondary closure leakage shall be determined once each quarter, or as soon as practicable after an unanticipated increase in pressurization gas flow is alarmed.

SR 5.2.16.a shall be implemented per ISI Criterion A.

b) The instrumentation monitoring PCRV penetration closure interspace pressurization gas flows, including alarms and high flow isolation, shall be functionally tested monthly and calibrated annually.

c) The instrumentation which monitors or alarms pressure in the core support floor and core support floor columns shall be functionally tested and calibrated annually.

SR 5.2.16.c shall be implemented per ISI Criterion A.

d) The controls, position indication, and fail safe operation for remote manual isolation valves associated with pressurizing, purging, and venting PCRV closures shall be functionally tested at five calendar year intervals, and for automatic isolation valves, annually, or at the next scheduled plant shutdown if these valves have not been tested during the previous year.

SR 5.2.16.d shall be implemented per ISI Criterion B.

e) The check valves on the HTFA purge lines shall be tested at five calendar year intervals.

SR 5.2.16.e shall be implemented per ISI Criterion B.

f) The check valves which are part of the HTFA or refueling penetrations shall only be tested when such a penetration is open for refueling or maintenance, if the check valves have not been tested in the last five years.

SR 5.2.16.f shall be implemented per ISI Criterion B.

<u>Basis for Specification SR 5.2.16</u>

The interval specified for determining the actual primary and secondary closure leakage is adequate to assure compliance with LCO 4.2.9.

In the determination of closure leakage at the reference differential pressure, laminar leakage flow shall be conservatively assumed, therefore in correcting the determined closure leakage to reference differential pressure, the ratio of the reference differential pressure, and test differential pressure shall be used.

The interval specified for functional testing and calibration of the instrumentation and alarms monitoring the penetration closure interspace pressurization gas flow will assure sensing and alarming any change in pressurization gas flow.

The interval specified for functional test and calibration of the instrumentation and alarms monitoring the core support floor and columns will assure sensing and alarming any change in their structural integrity.

The interval specified for valve testing is adequate to assure proper valve operation when isolation of the closure auxiliary piping is required.

Specification SR 5.2.17 - Helium Circulator Pelton Wheels

DELETE SPECIFICATION SR 5.2.17 IN ITS ENTIRETY

Specification SR 5.2.18 - Helium Circulators Surveillance

a) At the time of the first main turbine generator overhaul, one helium circulator unit shall be removed in its entirety from the PCRV and thoroughly inspected for signs of abnormal wear or component degradation.

   1) Such inspection shall include examination of bearing surfaces, seal surfaces, brake system, buffer seal system, and labyrinth seals.

2) The helium circulator compressor wheel rotor, turbine wheel, and Pelton wheel shall be inspected for both surface and subsurface defects in accordance with the appropriate methods, procedures, and associated acceptance criteria specified for Class I components in Article NB-2500, Section III, ASME Code.

b) Following the first complete helium circulator inspection, a previously uninspected helium circulator shal[illegible] be removed and inspected at ten calendar year intervals. The helium circulator compressor wheel rotor, turbine wheel, and Pelton wheel shall be inspected as specified in Paragraph a.2. Other helium circulator components, accessible without further disassembly than required to inspect these wheels, shall be visually examined.

Results of these examinations shall be submitted to the NRC staff for review and shall be evaluated to determine the need for scheduling additional future inspections.

SR 5.2.18 shall be implemented per ISI Criterion D.

<u>Basis for Specificat [illegible] SR 5.2.18</u>

Experience with the operation of single stage steam turbines as prime movers is common throughout industry.

Once such a machine is running satisfactorily, little or no wear occurs to it.

Unlike most designs of emergency systems of conventional nuclear power plants, the components of the Safe Shutdown System of the Fort St. Vrain plant are utilized and operated during normal operation of the plant. This includes the helium circulators.

The performance of the helium circulators is monitored during operation, i.e., instruments are provided with the capability to measure compressor differential pressure and flow, bearing temperature, bearing water temperature and flow, buffer helium flow, and shaft speed and vibration.

Examination at the time of the first turbine generator overhaul, and at approximately ten year intervals thereafter, is sufficient to monitor the condition of the helium circulator. The first turbine generator "tear-down" or overhaul usually occurs after one year running to check the total assembly. Only checks of components are performed during subsequent turbine generator overhauls.

The helium compressor and steam turbine blading should experience minimal wear in its running environment, and, with this length of service before inspection, will have undergone sufficient stress cycling to accurately indicate service life.

Specification SR 5.2.19 - IACM Diesel-Driven Pumps Surveillance

DELETE SPECIFICATION SR 5.2.19 IN ITS ENTIRETY

Specification SR 5.2.20 - ACM Diesel Driven Generator Surveillance

a) The diesel driven ACM generator shall be checked weekly by starting, and obtaining design speed and voltage.

b) The generator shall be tested monthly under load for a minimum of two hours. The load under this condition shall be at least 100% of design ACM equipment full load.

Basis for Specification SR 5.2.20

A weekly check of the Alternate Cooling Method generator to demonstrate its capability to start and a monthly test of the generator under load provides adequate assurance that the Alternate Cooling Method generator will be available to supply electrical power under the highly degraded, loss of forced circulation situation.

Specification SR 5.2.21 - Hand Valve and Transfer Switch Surveillance

Those pneumatically and electrically operated valves and electrical transfer switches that must be manually positioned to implement the ACM shall be tested twice annually at an interval between tests to be not less than four (4) months, nor greater than eight (8) months.

Basis for Specification SR 5.2.21

In the event that the ACM must be implemented, it is necessary to position pneumatically and electrically operated valves manually and to reposition electrical transfer switches. The test frequency and interval specified will assure operability in the event such operation is required.

Specification SR 5.2.22 - PGX Graphite Surveillance

PGX graphite surveillance specimens shall be installed into five (5) bottom transition reflector elements of the Fort St. Vrain core to provide a means for assessing the condition of the PGX graphite support blocks during operation of the reactor. These specimens (16 per reflector element) will be installed in reflector elements as indicated in Table 1 and will be removed at subsequent refueling intervals, as indicated in Table 1, unless the progressive examination of the specimens dictate otherwise.

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

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

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

Basis for Specification SR 5.2.22

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

SR 5.2.22 PGX GRAPHITE SURVEILLANCE

Table 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

*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.

Specification SR 5.2.23 - Firewater Booster Pump Surveillance

Each firewater booster pump shall be tested annually by providing motive power to one water turbine drive in conjunction with the performance of SR 5.2.7. In addition each pump shall be functionally tested quarterly. The associated instruments and controls shall functionally be tested quarterly and calibrated annually.

Basis for Specification SR 5.2.23

During accident conditions described in Final Safety Analysis Report, Section 14.4.2.1, one of the firewater booster pumps and one firewater pump are required to provide adequate core cooling. The specified testing interval is sufficient to ensure proper operation of the pump and associated controls.

Specification SR 5.2.24 - Circulating Water Makeup System Surveillance

The circulating water makeup system shall be verified operable as follows:

a) The circulating water makeup pond minimum inventory shall be verified daily. The pond level instrumentation shall be functionally tested monthly and calibrated annually.

b) The circulating water makeup pumps shall be functionally tested weekly. The pump controls and instrumentation including the fire water pump pits shall be functionally tested monthly and calibrated annually.

c) The valve lineup of the flow path between the circulating water storage ponds and the fire water pump pits shall be verified correct monthly.

Basis for Specification SR 5.2.24

The circulating water makeup system is required to supply water for fire suppression and safe shutdown cooling. The specified testing interval is sufficient to ensure proper operation of the pumps and controls. The system routinely operates during normal plant operation.

Specification SR 5.2.25 - Core Support Block Surveillance

The top surface of the core support block for fuel regions fitted with PGX graphite specimens shall be visually examined by remote TV for indication of cracks, in particular in areas where analysis shows the highest tensile stresses exist, at the refueling shutdown when the PGX graphite specimens are scheduled to be removed from the core in accordance with Technical Specification SR 5.2.22.

SR 5.2.25 shall be implemented per ISI Criterion D.

Basis for Specification SR 5.2.25

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

Specification SR 5.2.26 - Region Constraint Devices Surveillance

The region constraint devices (RCD's) shall be inspected at each refueling outage using the fuel handling machine from those regions being refueled as follows:

a) The upper core plenum shall be visually examined by remote TV to verify that RCD's within visible range are in place on top of the core.

b) As RCD's are removed, the fuel handling machine location coordinates and lifting force shall be monitored to verify that the RCD pins were engaged in the fuel columns and that they disengage as expected.

c) Selected RCD's shall be visually examined by remote TV in the fuel handling machine after removal to verify their structural integrity.

d) As RCD's are re-installed, the fuel handling machine location coordinates shall be monitored to verify that the RCD pins have engaged in the fuel columns.

SR 5.2.26 shall be implemented per ISI Criterion B.

Basis for Specification SR 5.2.26

Region constraint devices, located on top of fuel columns of generally three adjacent fuel regions, restrain region movements in relation to one another by means of centering pins inserted in the handling hole of the upper plenum elements.

Visual examination of the upper core plenum and comparison of the as-installed/as-found RCD coordinates will assure that the RCD's remain in place and that no phenomenon is occurring which could cause them to disengage from the fuel columns. Comparison of RCD coordinates will require correction to account for changes in fuel column height due to irradiation of graphite and coordinate changes which will occur when RCD's are removed from a different refueling penetration than the one from which they were installed.

Monitoring the lifting force to remove the RCD's 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. Removal and re-installation will act as go/no-go dimensional test of the region constraint devices.

Visually examining and photographing selected RCD's in the fuel handling machine will assure that there are no unacceptable deformations, loose or missing parts, or other visible defects.

<u>Specification SR 5.2.27 - Helium Shutoff Valves Surveillance</u>

Proper closure of the helium shutoff valves shall be monitored annually, or at the next scheduled plant shutdown, if such monitoring has not been performed during the previous year.

SR 5.2.27 shall be implemented per ISI Criterion C.

Basis for Specification SR 5.2.27

The helium shutoff valves are self-actuated check valves which close when the corresponding circulators are shutdown or tripped. Simultaneous long term failure of both the circulator and its helium shutoff valve, under very degraded conditions of remaining plant equipment, could lead to a situation analogous to a loss of forced circulation accident, due to the open recirculation path between circulator outlet and inlet plenums.

Verification that the helium shutoff valves close properly will provide assurance that the residual heat removal capability would not be degraded by the malfunction of a helium shutoff valve.

Specification SR 5.2.28 - PCRV Penetrations and Closures Surveillance

a) Accessible portions of PCRV penetration pressure retaining welds shall be examined for indications of surface defects as follows:

   1) Surface examine (MT or PT) the following three welds in one steam generator penetration in each loop at five calendar year intervals:

      - the penetration shell to secondary closure weld,

      - the secondary closure to upper bellows support weld, and

- the lower bellows support to reheat header sleeve weld.

2) Surface examine (MT or PT) the following two welds in the bottom access penetration at 10 calendar year intervals:

- the penetration shell to spherical head weld, and

- the spherical head to closure flange weld.

SR 5.2.28.a shall be implemented per ISI Criterion C.

b) Accessible portions of the PCRV penetration closure and flow restrictor restraint components shall be examined for indications of defects as follows:

1) Visually examine the helium circulator restraint system (cylinder, ring, and bolting) for one penetration in each loop at five calendar year intervals.

SR 5.2.28.b.1 shall be implemented per ISI Criterion C.

2) Visually examine the refueling penetration holddown plate bolting at each refueling outage.

   SR 5.2.28.b.2 shall be implemented per ISI Criterion B.

3) Visually examine the bottom access penetration primary closure split ring assembly and its secondary closure bolting at 10 calendar year intervals.

   SR 5.2.28.b.3 shall be implemented per ISI Criterion C.

c) Accessible portions of the PCRV safety valve penetration containment tank support components shall be examined at 10 calendar year intervals for indications of defects as follows:

1) Surface examine (MT or PT) the support skirt to tank attachment weld.

2) Visually examine the support skirt between the tank and PCRV outer wall.

3) Visually examine, torque, and tension test the bolting attaching the support skirt to the PCRV outer wall.

SR 5.2.28.c shall be implemented per ISI Criterion C.

Basis for Specification SR 5.2.28

Structural integrity of Fort St. Vrain PCRV penetration secondary pressure retaining boundaries is normally verified by continuous leakage monitoring and by periodic leakage testing of the penetration interspace. The specified examinations of accessible circumferential welds at structural discontinuities will provide additional assurance concerning the continued integrity of the secondary pressure boundary at these critical locations.

Examination of accessible penetration closures, flow restrictors, and equipment restraint or support components provides assurance that these components remain structurally sound and capable of performing their safety function under both normal and accident conditions.

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5.3 SECONDARY COOLANT SYSTEM - SURVEILLANCE REQUIREMENTS

Applicability

Applies to the surveillance of the secondary (steam) coolant system, including the steam generators and turbine plant.

Objective

To ensure the core cooling capability of the components of the steam plant system.

Specification SR 5.3.1 - Steam/Water Dump System Surveillance

a) The steam/water dump valves shall be tested individually every three months.

b) The steam/water dump tank level indicators shall be checked daily, and functionally tested every three months.

SR 5.3.1.b shall be implemented per ISI Criterion A.

c) The steam/water dump tank level, pressure and temperature instruments (including indicators, alarms, and interlocks - where applicable) shall be functionally tested and calibrated annually, or at the next scheduled plant shutdown if such surveillance has not been performed during the previous year.

SR 5.3.1.c shall be implemented per ISI Criterion B.

Basis for Specification SR 5.3.1

The steam/water dump system is provided to minimize water in-leakage into the core as a result of a steam generator tube rupture (FSAR, Section 6.3). Satisfactory operation of the dump valves, as is sufficiently demonstrated by testing every three months, will minimize core damage and primary coolant system pressure rise in the event of a steam generator tube rupture.

The dump valve test will be accomplished by closing the (normally locked open) block valve downstream of the dump valve to be tested. After operation of the dump valve, the block valve will again be locked open, returning the dump valve to service.

The specified frequency for instrumentation functional test and calibration is adequate to assure that the water level in the steam/water dump tank does not exceed the limits of LCO 4.3.3, and, in case of dump, to confirm that the proper steam generator has been dumped, and to prevent venting and draining of the tank to the radioactive gaseous and liquid systems before the contents have been adequately cooled.

Specification SR 5.3.2 - Main and Hot Reheat Steam Stop Check Valves Surveillance

The main steam and hot reheat steam stop check valves shall be full stroke tested in accordance with specification SR 5.3.4 and partial stroke tested once per week.

SR 5.3.2 shall be implemented per ISI Criterion A.

Basis for Specification SR 5.3.2

The main steam stop check and hot reheat stop check valves will be partially stroked once a week during plant operation. Full stroking tests are impractical because complete closure of any one valve would automatically shut down one or more circulators. Therefore, the valves will be stroked during power operation by means of special electrical circuitry in the hydraulic control system which limits closure to 10% without interfering with emergency closure action called for by the plant protective system. This test will demonstrate that the valves are free to close when required, without causing severe pressure, temperature, flow, or power generation transients.

Specification SR 5.3.3 - Bypass and Pressure Relief Valves Surveillance

The main steam and hot reheat steam power operated (electromatic) pressure relief valves, and the six hot reheat steam bypass valves shall be tested once per year, or at the next scheduled plant shutdown if the valves have not been tested during the previous year. The main steam bypass valves shall be tested in accordance with specification SR 5.3.4.

SR 5.3.3 shall be implemented per ISI Criterion A.

Basis for Specification SR 5.3.3

The specified secondary (steam) coolant system bypass valves and pressure relief valves will be tested during plant shutdown as follows:

a) The main steam and hot reheat steam power operated pressure relief valves will be tested by exercising the valves.

b) The main steam bypass valves will be tested for operability by cycling the valves.

c) The six hot reheat steam bypass valves will be tested by exercising each valve to ensure freedom of movement.

The main steam bypass valves divert up to 77% steam flow (via desuperheaters) to the bypass flash tank on turbine trip or loop isolation, so that the steam is available for driving helium circulators, boiler feedpump turbines, etc. The main steam power operated relief valves divert the remaining steam flow to atmosphere.

The six hot reheat steam bypass valves and the power operated pressure relief valves ensure a continuous steam flow path from the helium circulators for decay heat removal.

The tests required on the above valves will demonstrate that each valve will function properly. Test frequency is considered adequate for assuring valve operability at all times.

<u>Specification SR 5.3.4 - Safe Shutdown Cooling Valves, Surveillance</u>

Those valves that are pneumatically, hydraulically, or electrically operated, that are required for actuation of the Safe Shutdown Cooling mode of operation, shall be tested annually, or at the next scheduled plant shutdown if these valves have not been tested during the previous year.

In addition, the above test shall include the normally closed check valves which are required to open for actuation of the safe shutdown cooling mode of operation, when such testing is practical.

SR 5.3.4 shall be implemented per ISI Criterion B.

Basis for Specification SR 5.3.4

The safe shutdown cooling mode of operation utilizes systems or portions of systems that are in use during normal plant operation. In many cases, those valves required to initiate Safe Shutdown Cooling are not called upon to function during normal operation of the plant, except to stand fully closed or open.

Testing of these valves will assure their operation if called upon to initiate the Safe Shutdown Cooling mode of operation.

During reactor operation, the instrumentation required to monitor and control the Safe-Shutdown mode of cooling is normally in use and any malfunction would be immediately brought to the attention of the operator. That instrumentation not normally in use is tested at intervals specified by other surveillance requirements in this Technical Specification.

Safe Shutdown Cooling, the systems or portions of systems involved, are discussed in Sections 10.3.9 and 10.3.10 of the FSAR and are represented in FSAR, Figure 10.3-4.

Valve testing will include, as applicable, full stroking each valve, or an observation that the valve disc travels from the valve normal operating position to the position required to perform the safety function, an observation that the remote position indicators accurately reflect actual valve position, and a measurement of the full stroke time for the hydraulically actuated automatic valves.

Specification SR 5.3.5 - Hydraulic Power System Surveillance

The pressure indicators and low pressure alarms on the hydraulic oil accumulators pressurizing gas and on the hydraulic power supply lines shall be functionally tested once every three months and calibrated once per year.

Basis for Specification SR 5.3.5

The hydraulic power system is a normally operating system. Malfunctions in this system will normally be detected by failure of the hydraulic oil pumps or hydraulic oil accumulators to maintain a supply of hydraulic oil at or above 2500 psig. Functional tests and calibrations of the pressure indicators and low pressure alarms on the above basis will assure the actuation of these alarms upon a malfunction of the hydraulic power system which may compromise the capability of operating critical valves.

Specification SR 5.3.6 - Instrument Air System Surveillance

The pressure indicators and low pressure alarms on the instrument air receiver tanks and headers shall be functionally tested monthly and calibrated annually.

Basis for Specification SR 5.3.6

The instrument air system is a normally operating system. Malfunctions in this system will be normally detected by failure of the instrument air compressors to maintain the instrument air receiver tanks at a pressure above the alarm setpoint. Functional tests of the pressure indicators and low pressure alarms on a monthly basis and calibration on an annual basis will assure the actuation of these alarms upon a malfunction of the instrument air system which may compromise the capability of operating critical valves.

Specification SR 5.3.7 - Secondary Coolant Activity Surveillance

The secondary coolant system will be analyzed for ^{131}I, tritium, and gross beta plus gamma concentration once per week during reactor operation.

If the secondary coolant activity level reaches 25% of the limit of LCO 4.3.8, or the secondary coolant activity level increases by a factor of 25% over the previous equilibrium value at the same reactor power level, the frequency of sampling and analysis shall be increased to a minimum of once each day until the activity level decreases or reaches a new equilibrium value (defined by four consecutive daily analyses whose results are within ±10%), at which time weekly sampling may be resumed.

Basis for Specification SR 5.3.7

The specification surveillance interval is adequate to monitor the activity of the secondary coolant.

Specification SR 5.3.8 - Hydraulic Snubbers Surveillance

The following surveillance requirements apply to all Class I piping system hydraulic snubbers:

a) All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and mechanical linkage connections to the piping and anchor to verify snubber operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 Months ± 25%
1	12 Months ± 25%
2	6 Months ± 25%
3, 4	124 Days ± 25%
5, 6, 7	62 Days ± 25%
≥ 8	31 Days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

b) All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

c) The initial inspection shall be performed within 6 months from issuance of this Technical Specification. For the purpose of entering the schedule in a) above, it shall be assumed that the facility had been on a six (6) month inspection interval.

d) Once each refueling cycle, starting with the first refueling, a representative sample of 10 hydraulic snubbers or approximately 10 percent of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional 10 percent or ten snubbers shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capacity greater than 50,000 pounds need not be functionally tested.

Basis for Specification SR 5.3.8

All Class I hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

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 such early inspection performed before the original required time interval has elapsed (nominal time less 25 percent) may not be used to lengthen the required inspection interval. Any inspection where results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests, or analysis, to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Snubbers in high radiation areas or those especially difficult to remove need not be selected for functional tests provided operability was previously verified.

Specification SR 5.3.9 - Safety Valves Surveillance

The steam generator superheater and reheater safety valves and the steam/water dump tank safety valves shall be tested at five calendar year intervals to verify their setpoint.

SR 5.3.9 shall be implemented per ISI Criterion B.

Basis for Specification SR 5.3.9

The safety valves protect the integrity of the steam generators, which are part of the reactor coolant boundary, and of the dump tank, which may contain radioactive fluids. Testing the safety valve setpoints will assure that the pressure within the equipment remains within design limits.

When practical, testing of the safety valves will be scheduled during the surveillance interval so that testing of one (or more) safety valve(s) of similar type and operating conditions several times during the interval will provide additional confidence in safety valve reliability and adequate overpressure protection.

Specification SR 5.3.10 - Secondary Coolant System Instrumentation Surveillance

The secondary coolant reheat steam instrumentation used

a) for control and indication of emergency condensate flow to the reheaters and reheater backpressure, in case of safe shutdown cooling,

b) to automatically open the reheater discharge bypass on high pressure, and

c) to monitor reheater discharge bypass temperature, and reheater inlet temperature,

shall be functionally tested and calibrated annually, or at the next scheduled plant shutdown if such surveillance was not performed during the previous year.

SR 5.3.10 shall be implemented per ISI Criterion B.

Basis for Specification SR 5.3.10

The frequency specified for surveillance of the above instrumentation will assure that they perform their expected automatic actions, and that the operator will be provided with accurate information which he can use for safe shutdown cooling or to avoid abnormal equipment operation.

Specification 5.3.11 - Steam Generator Bimetallic Welds Surveillance

The accessible portions of steam generator bimetallic welds shall be volumetrically examined for indications of subsurface defects as follows:

a) The main steam ring header collector to main steam piping weld for one steam generator module in each loop at five calendar year intervals.

b) The main steam ring header collector to collector drain piping weld for one steam generator module in each loop at five calendar year intervals.

c) The same two steam generator modules initially selected shall be re-examined at each interval.

d) The bimetallic welds described in a) and b) shall also be inspected for two other steam generator modules in each loop during the initial examination.

SR 5.3.11 shall be implemented per ISI Criterion C.

Basis for Specification 5.3.11

The steam generator crossover tube bimetallic welds between Incoloy 800 and 2 1/4 Cr-1 Mo materials are not accessible for examination. The bimetallic welds between the steam generator ring header collector, the main steam piping, and the collector drain piping are accessible, involve the same materials and operate at conditions not significantly different from the crossover tube bimetallic welds. The collector drain piping weld is also geometrically similar to the crossover tube weld. Examination of selected bimetallic welds that are accessible will provide additional assurance concerning the continued integrity of steam generator bimetallic welds. Although no 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 first examination to establish a baseline which could be used, should defects be found in later inspections and additional examinations subsequently be required.

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