

ATTACHMENT 2

PROPOSED MODIFICATIONS TO THE FORT ST. VRAIN  
TECHNICAL SPECIFICATIONS

(SECTION 2)

#### 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  $\pm 25\%$  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,

$$T_{avg} = \frac{\sum_{i=1}^{37} T_i}{37}$$

where  $T_i$  = individual refueling region outlet temperature.

ATTACHMENT 3

PROPOSED MODIFICATIONS TO THE  
FORT ST. VRAIN TECHNICAL SPECIFICATIONS

(SECTION 5 - INCLUDING IN-SERVICE INSPECTION CHANGES)



## 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. \_\_\_\_ 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. \_\_\_\_ 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. \_\_\_\_ 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. \_\_\_\_ 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  $0.01 \Delta k$ , 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  $0.01 \Delta k$  would be unexpected and its occurrence would be thoroughly investigated and evaluated. The value of  $0.01 \Delta k$  is considered to be a safe limit since a shutdown margin of at least  $0.01 \Delta k$  with the highest worth rod pair fully withdrawn is always maintained (see 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 (+5) effective days at

rated thermal power  
after refueling.

- 3) At the equivalent of 40  
(+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 - PCRVR and PCRVR Penetration Overpressure Protection Surveillance

a) Each of the two overpressure protection assemblies protecting the PCRVR 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 PCRVR 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 PCR 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.

Basis for Specification SR 5.2.2

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 at ten calendar year intervals thereafter.

| b) PCRV deformations and deflections at vessel midheight  
| and at the center of the top head shall be monitored  
| at five calendar year intervals during a vessel  
| pressurization to operating pressure.

| 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.

Basis for Specification SR 5.2.4

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 PCR/V response.

Further discussion on the significance of concrete cracks in the PCR/V is given in Section 5.12.5 of the FSAR.

Observed crack development with time during reactor operation will be related to the PCR/V structural response as monitored by the installed sensors and deflection measurements. Details of the PCR/V 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 PCR/V deformations and deflections is  
| the best indication of PCR/V structural performance and

| verifies that the PCRVR response is elastic and that no  
| significant permanent strains exist.

| Visual examination of the PCRVR 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 PCRVR.

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 PCRVR 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 PCRVR 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  $^{90}\text{Sr}$  inventory in the reactor circuit. The probes removed shall also be analyzed for  $^{131}\text{I}$ .

#### 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  $^{90}\text{Sr}$  and the results shall be used to establish the total  $^{90}\text{Sr}$  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  $^{90}\text{Sr}$  which were made between probe removals. The analysis for  $^{131}\text{I}$  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 batter-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.

Basis for Specification SR 5.2.10

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 LCD 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 analysis whose results are within  $\pm 10\%$ ) 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 analysis 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 4.2.15 - 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.

Basis for Specification SR 5.2.16

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 shall 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.

Basis for Specification SR 5.2.18

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 circulator 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.

| Specification SR 5.2.27 - Helium Shutoff Valves  
| Surveillance

| 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 PCRIV 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 PCRIV outer wall.

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

Specification SR 5.3.4 - Safe Shutdown Cooling Valves, Surveillance

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 values.

Specification SR 5.3.7 - Secondary Coolant Activity  
Surveillance

The secondary coolant system will be analyzed for  $^{131}\text{I}$ , 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 analysis whose results are within  $\pm 10\%$ ), 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:

<u>Number of Snubbers Found Inoperable During Inspection or During Inspection Interval</u>	<u>Next Required Inspection Interval</u>
0	18 Months $\pm$ 25%
1	12 Months $\pm$ 25%
2	6 Months $\pm$ 25%
3, 4	124 Days $\pm$ 25%
5, 6, 7	62 Days $\pm$ 25%
$\geq 8$	31 Days $\pm$ 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 or 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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5.4 INSTRUMENTATION AND CONTROL SYSTEMS - SURVEILLANCE AND  
CALIBRATION REQUIREMENTS

Applicability

Applies to the surveillance and calibration of the reactor protective system and other critical instrumentation and controls.

Objective

To assure the operability of the reactor protection system and other critical instrumentation and controls by specifying their surveillance and calibration frequencies.

Specification SR 5.4.1 - Reactor Protective System and  
Other Critical Instrumentation and Control Checks,  
Calibrations, and Tests

The surveillance and calibration tests of the protective instrumentation shall be as given in Tables 5.4.1 through 5.4.4:

- a) Table 5.4.1 - Minimum Frequencies for checks, calibrations, and testing of scram system.
- b) Table 5.4.2 - Minimum Frequencies for checks, calibrations, and testing of Loop Shutdown System.
- c) Table 5.4.3 - Minimum Frequencies for checks, calibrations, and testing of Circulator Trip System.

- d) Table 5.4.4 - Minimum Frequencies for checks, calibrations, and testing of Rod Withdrawal Prohibit System.

Basis for Specification SR 5.4.1

The specified surveillance check and test minimum frequencies are based on established industry practice and operating experience at conventional and nuclear power plants. The testing is in accordance with the IEEE Criteria for Nuclear Power Plant Protection Systems, and in accordance with accepted industry standards.

Calibration frequency of the instrument channels listed in Tables 5.4.1, 5.4.2, 5.4.3, 5.4.4 are divided into three categories: passive type indicating devices that can be compared with like units on a continuous basis; semiconductor devices and detectors that may drift or lose sensitivity; and on-off sensors which must be tripped by an external source to determine their setpoint. Drift tests by GGA on transducers similar to the reactor pressure transducers (FSAR Section 7.3.3.2) indicate insignificant long term drift. Therefore a once per refueling cycle calibration was selected for passive devices (thermo-couples, pressure transducers, etc.). Devices incorporating semiconductors, particularly amplifiers, will be also calibrated on a once per refueling cycle basis, and any drift in response or bistable setpoint will be discovered from the test

program. Drift of electronic apparatus is not the only consideration in determining a calibration frequency; for example, the change in power distribution and loss of detector chamber sensitivity require that the nuclear power range system be calibrated every month. On-off sensors are calibrated and tested on a once per refueling cycle basis.

TABLE 5.4-1

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

Channel Description	Function	Frequency(1)	Method
1. Manual (Control Room)	a. Test	R	a. Manually trip system.
2. Manual (I-49)	a. Test	M	a. Manually trip each channel.
3. Startup Channel	a. Check	D	a. Comparison of two separate channel indicators.
	b. Test	P	b. Internal test signal to verify trips and alarms.
	c. Calibrate	R	c. Internal test signal shall be checked and calibrated to assure that its output is in accordance with the design requirements. This shall be done after completing the external test signal procedure by checking the output indication when turning the internal test signal switch.
4. Linear Power Channel	a. Check	D	a. Comparison of six separate channel indicators.
	b. Test	M	b. Internal test signal to verify trips and alarms.
	c. Calibrate	D	c. Channel adjusted to agree with heat balance calculation.
	d. Calibrate	R	d. Internal test signals to adjust trips and indications.
5. Wide Range Power Channel	a. Check	D	a. Comparison of three separate indicators.



TABLE 5.4-1

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

Channel Description	Function	Frequency(1)	Method
5. (Cont'd)	b. Test	P	b. Internal test signals to verify trips and alarms.
	c. Calibrate	M	c. Channel adjusted to agree with heat balance calculation.
	d. Calibrate	R	d. Internal test signals to adjust trips and indications.
6. Primary Coolant Moisture (All Channels)	a. Check	D	a. Comparison of two separate high level channel mirror temperature indications.
	b. Check	D	b. Comparison of six separate low level channel mirror temperature indications.
	c. Calibrate	R	c. Inject moisture laden gas into sample lines.
	d. Check	D	d. Verification of eight separate monitor's sample flow, per Item (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.

TABLE 5.4-1

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

Channel Description	Function	Frequency(1)	Method
8. (Cont'd)	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 to an NBS traceable standard. 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 each thermocouple output to an NBS traceable standard. Internal test signal to adjust trips and indicators.
11. Hot Reheat Header Pressure	a. Test	M	a. Reduce pressure at sensor to trip channel, verify alarms and indications.

TABLE 5.4-1

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

Channel Description	Function	Frequency(1)	Method
11. (Cont'd)	b. Calibrate	R	b. Known pressure applied at sensor to adjust trips.
12. Main Steam Pressure	a. Test	M	a. Reduce pressure at sensor to trip channel, verify alarms and indications.
	b. Calibrate	R	b. Known pressure applied at sensor to adjust trips.
13. Two Loop Trouble	a. Test	M	a. Special test module used to trip channel by energizing each of four appropriate pairs of two-loop trouble relays.
	b. Test	R	b. Trip logic to cause two loop trouble scram.
14. Plant 400 V Power Loss	a. Test	M	a. Trip each channel by applying simulated loss of voltage signal; verify alarms and indications.
15. High Reactor Building Temperature (Pipe Cavity)	a. Check	D	a. Comparison of three separate channel indicators.
	b. Test	M	b. Trip channel, verify alarms and indications. Internal test signal to verify trips and alarms.

TABLE 5.4-1

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

Channel Description	Function	Frequency(1)	Method
15. (Cont'd)	c. Calibrate	R	c. Compare each thermocouple output to an NBS traceable standard to adjust temperature trip point.

NOTE 1: D - Daily when in use.  
 M - Monthly.  
 R - Once per refueling cycle.  
 P - Prior to each startup, if not done the previous week.

TABLE 5.4-2

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

Channel Description	Function	Frequency(1)	Method
1. Steam Pipe Rupture (Pipe Cavity)	a. Check	D	a. Comparison of separate ultrasonic channel indicators/loop.
	b. Test	M	b. Pulse test one temperature and pressure channel with another temperature and pressure channel tripped, while simultaneously having two ultrasonic channels tripped.
	c.		c. DELETED
	d. Test	M	d. Pressure switch actuated by pressure applied at sensor.
	e. Test	M	e. Temperature switch actuated by heat applied at sensor.
	f. Test	M	f. Internal test signal to adjust ultrasonic trip.
	g. Test	M	g. Trip test signal solenoid valves to verify loop integrity.
	h. Calibrate	R	h. Known pressure applied at sensor to verify response.
	i. Calibrate	R	i. Known temperature applied at sensor to verify response
	j. Calibrate	R	j. Known sound applied at sensor to verify response.
2. Steam Pipe Rupture (Under PCRV)	a. Check	D	a. Comparison of separate ultrasonic channel indicators/loop.

TABLE 5.4-2

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

Channel Description	Function	Frequency(1)	Method
2. (Cont'd)	b. Test	M	b. Pulse test one temperature and pressure channel with another temperature and pressure channel tripped, while simultaneously having two ultrasonic channels tripped.
	c.		c. DELETED
	d. Test	M	d. Pressure switch actuated by pressure applied at sensor.
	e. Test	M	e. Temperature switch actuated by heat applied at sensor.
	f. Test	M	f. Internal test signal to adjust ultrasonic trip.
	g. Test	M	g. Trip test signal solenoid valves to verify loop integrity.
	h. Calibrate	R	h. Known pressure applied at sensor to verify response.
	i. Calibrate	R	i. Known temperature applied at sensor to verify response.
	j. Calibrate	R	j. Known sound applied at detector to verify response.
3. Circulator 1A and 1B Tripped	a. Test	M	a. Pulse test and verify proper indications.
	b. Test	R	b. Trip both circulators to test loop shutdown.
4. Circulator 1C and 1D Tripped	a. Test	M	a. Pulse test and verify proper indications.



TABLE 5.4-2

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

Channel Description	Function	Frequency(1)	Method
4. (Cont'd)	b. Test	R	b. Trip both circulators to test loop shutdown.
5. Steam Generator Penetration Pressure	a. Test	M	a. Pressure switches actuated by pressure applied.
	b. Test	M	b. Pulse test each channel with another channel tripped and verify proper indications.
	c. Calibrate	R	c. Known pressure applied at sensor to adjust trip.
6. Reheat Header Activity	a. Check	D	a. Comparison of three separate indicators in each loop.
	b. Test	M	b. Pulse test each channel with another channel tripped and verify proper indications.
	c. Calibrate	R	c. Expose sensor to known radiation source and adjust trips and indicators.
7. Superheat Header Temperature	a. Check	D	a. Comparison of three separate temperature indicator's per loop
	b. Check	D	b. Comparison of three separate temperature differential indicators.
	c. Test	M	c. Pulse test one channel with another channel tripped and verify proper indications.



TABLE 5.4-2

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

Channel Description	Function	Frequency(1)	Method
7. (Cont'd)	d. Calibrate	R	d. Compare each thermocouple output to an NBS traceable standard. Internal test signal to adjust trips and indicators.
8. Primary Coolant Moisture (Low Level Channels)	a. Test	M	a. Trip each channel, verify proper indications.
	b. Test	M	b. Trip each channel, pulse test other loop to check loop identification.
9. Primary Coolant Pressure	a. Test	M	a. Pulse test one channel with another channel tripped and verify proper indications, both channels.

NOTE 1: D - Daily when in use.  
 M - Monthly.  
 R - Once per refueling cycle.  
 Prior to each startup, if not done the previous week.

TABLE 5.4-3

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

Channel Description	Function	Frequency(1)	Method
1. Circulator Speed, Steam, and Water	a. Check	D	a. Comparison of six separate speed indications per circulator.
	b. Test	M	b. Internal test signal to verify trip setting and indicators.
	c. Test	M	c. Pulse test one channel with another channel tripped, and verify proper indications.
	d. Calibrate	R	d. Known pulse frequency applied at sensor to adjust trips and indicators.
2. Feedwater Flow	a. Check	D	a. Comparison of six separate indicators per loop.
	b. Test	M	b. Internal test signal to verify trip setting and indications.
	c. Test	M	c. Pulse test one channel with another channel tripped, and verify proper indications.
	d. Calibrate	R	d. Apply known $\Delta P$ at flow transmitter. Internal test signal to adjust trips and indicators.
3. Circulator Bearing Water Pressure	a. Check	D	a. Comparison of three separate indicators/circulator.
	b. Test	M	b. Pulse test one channel with another channel tripped and verify proper indications.
	c. Calibrate	R	c. Known pressure applied to adjust trip setting.

TABLE 5.4-3

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

Channel Description	Function	Frequency(1)	Method
4. Circulator Penetration Pressure	a. Test	M	a. Pressure switches actuated by pressure applied.
	b. Test	M	b. Pulse test one channel with another channel tripped and verify proper indications.
	c. Calibrate	R	c. Known pressure applied at sensor to adjust trip setting.
5. Circulator Drain Pressure	a. Check	D	a. Comparison of three separate indicators/circulator.
	b. Test	M	b. Pulse test one channel with another channel tripped and verify proper indications.
	c. Calibrate	R	c. Known pressure applied at sensor to adjust trip setting.
6. Circulator Seal Malfunction	a. Check	D	a. Comparison of three separate indicators/circulator.
	b. Test	M	b. Pulse test one channel with another channel tripped and verify proper indications.
	c. Calibrate	R	c. Known pressure applied at sensor to adjust trip setting.

TABLE 5.4-3

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

Channel Description	Function	Frequency(1)	Method
7. Circulator Trip (Manual)	a. Test	R	a. Trip steam turbine drives. Verify water turbine automatic start.

NOTE 1: D - Daily when in use.  
 M - Monthly.  
 R - Once per refueling cycle.  
 P - Prior to each startup, if not done the previous week.

TABLE 5.4-4

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

Channel Description	Function	Frequency(1)	Method
1. Startup Channel	a. Check	D	a. Comparison of two separate channel indicators.
	b. Test	P	b. Internal test signal to verify all trips and alarms.
	c. Calibrate	R	c. The internal test signal shall be checked and calibrated to assure that its output is in accordance with the design requirements. This shall be done after completing the external test signal procedure by checking the output indication when turning the internal test signal switch.
2. Linear Channel	a. Check	D	a. Comparison of six separate level indicators.
	b. Test	M	b. Internal test signal to verify trips and alarms.
	c. Calibrate	D	c. Channel adjusted to agree with heat balance calculation.
	d. Calibrate	R	d. Internal test signals to adjust trips and indications.
3. Wide Range Power Channel	a. Check	D	a. Comparison of three separate indicators.
	b. Test	P	b. Internal test signals to verify trips and alarms.

TABLE 5.4-4

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

Channel Description	Function	Frequency(1)	Method
3. (Cont'd)	c. Calibration	M	c. Channel adjusted to agree with heat balance calculation.
	d. Calibration	R	d. Internal test signals to adjust trips and indications.
4. Multiple Rod Pair Withdrawal	a. Test	P	a. Attempt two rod pair withdrawal.
	b. Check	R	b. Simulate current through sensor to verify trip and alarms.

NOTE 1: D - Daily when in use.  
 M - Monthly.  
 R - Once per refueling cycle.  
 P - Prior to each startup, if not done the previous week.

Specification SR 5.4.2 - Control Room Smoke Detector

The control room smoke detectors and alarms will be functionally tested once per year.

Basis for Specification SR 5.4.2

The control room smoke detectors provide for sensing of the smoke in the outlet air ducts from both the control room and the auxiliary electrical room. In the event of any fire or smoke in the control panels, alarms will be initiated.

Specification SR 5.4.3 - Core Region Outlet Temperature Instrumentation

The output of two thermocouples measuring each region outlet temperature will be checked daily during power operation. If the indicated temperatures for a region differ by  $\geq \pm 75^{\circ}\text{F}$ , a calibration shall be made and the faulty thermocouple replaced by an operable thermocouple. The core region outlet thermocouple shall be calibrated once per year during power operation by traversing a calibrated thermocouple along each of the seven coolant thermocouple assemblies.



Basis for Specification 5.4.3

The long-term thermocouple drift is estimated to be  $\leq 15^{\circ}\text{F}$  per year and this drift was included in the measurement uncertainty of  $\pm 50^{\circ}\text{F}$  used to establish LCO 4.1.7. With this measurement uncertainty, a root mean square difference of  $\geq \pm 75^{\circ}\text{F}$  would be an indication of a faulty reading. Daily checks and yearly calibrations are considered adequate since the expected drift in calibration is small and has been included in establishing LCO 4.1.7 (See FSAR Section 7.3.3).

Specification SR 5.4.4 - PCR/V Cooling Water System  
Temperature Scanner - Surveillance Requirement

PCR/V Cooling System temperature scanner readings shall be checked by comparison of representative liner cooling tube thermocouple outputs to their respective subheader temperatures and associated alarms tested once per month during power operation.

All thirty-six (36) outlet subheader temperature indicators shall be calibrated annually. In addition, ninety-seven (97) liner cooling tube outlet thermocouples shall be calibrated annually.

Basis for Specification SR 5.4.4

The temperature scanner for the PCRV cooling system provides for continuous temperature monitoring of the outlet water temperature of each individual liner cooling tube and alarming of high outlet temperatures.

The surveillance interval specified is sufficient to detect any drift in the output of the individual thermocouples or scanner electronics to assure the temperature limitations of the PCRV cooling system are not exceeded.

The ninety-seven (97) thermocouples shall be distributed among the thirty-six (36) subheaders so that between 16.7% and 21.5% of the total in each subheader are calibrated each year. Thus, the maximum time between calibration of any one thermocouple, or any complete subheader, shall not exceed six (6) years. The overall percentage of thermocouples calibrated per year exceeds 18%.

The surveillance interval for calibration, combined with that for checking, assures sufficient accuracy of temperature measurement to adequately protect the PCRV concrete.

Specification SR 5.4.5 - PCRV Cooling Water System Flow  
Scanner - Surveillance Requirement

A PCRV Cooling System flow scanner readout shall be taken and alarms functionally checked monthly. The scanner and alarms, and six (6) subheader flow meters shall be calibrated annually.

Basis for Specification SR 5.4.5

The flow scanner acts as a backup to the temperature scanner and initiates no automatic protective action, only an alarm. Because a restriction or a leak in the system would develop over a period of time, the monthly interval for comparing scanner readouts is sufficient to detect any long term change in the system.

Specification SR 5.4.6 - Core  $\Delta P$  Indicator - Surveillance Requirement

The core  $\Delta P$  instrumentation shall be calibrated on a once per refueling cycle interval.

Basis for Specification SR 5.4.6

Core differential pressure is an indication of gross blockage of flow in the core.

Specification SR 5.4.7 - Control Room Temperature -  
Surveillance Requirement

The control room temperature control thermostat shall be functionally tested monthly and calibrated annually.

Basis for Specification SR 5.4.7

The surveillance interval specified for functional testing and calibration of the control room thermostat will assure its ability to not only control the room temperature as desired, but to also indicate the correct room temperature within the accuracy of the instrument.

Specification SR 5.4.8 - Power to Flow Instrumentation -  
Surveillance Requirement

The power to flow indication shall be verified daily and shall be calibrated once per refueling cycle.

Basis for Specification SR 5.4.8

The power to flow ratio indication is an indication of the balance between the heat generation and removal within the primary coolant system. A verification of the power to flow indication on a daily basis is adequate to assure the instrument is indicating properly. In addition, any change in reactor power level no matter how small, should produce a change in the power to flow ratio indication. A lack of response by this instrumentation would be noticed by the operator. Calibration of the instrumentation on a

once per refueling cycle basis is acceptable by industry standards for this type of instrumentation.

Specification SR 5.4.9 - Area and Miscellaneous Process  
Radiation Monitors - Surveillance Requirement

The area radiation monitors shall be functionally checked weekly and calibrated annually.

Basis for Specification SR 5.4.9

The surveillance interval specified for functional testing and calibration are adequate to assure the proper operation of these detectors.

Specification SR 5.4.10 - Seismic Instrumentation -  
Surveillance Requirement

The Seismic Instrumentation shall be functionally tested every six months and calibrated every two years.

Basis for Specification 5.4.10

The intervals specified for testing and calibration of the Seismic Instrumentation are recommended by the manufacturer to assure the instruments operate as intended.

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.

Specification      SR 5.4.13 - 480 V      Switchgear      Room

Temperature Indication - Surveillance Requirement

The 480 V switchgear room temperature indicator and alarm shall be functionally tested monthly and calibrated annually.

Basis for Specification SR 5.4.13

The surveillance interval specified for this instrumentation assures its proper operation on a continuous basis.



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## 5.5 CONFINEMENT SYSTEM - SURVEILLANCE REQUIREMENTS

### Applicability

Applies to the surveillance of the reactor building (confinement) and the reactor building ventilation system.

### Objective

To ensure that the structure and components of the reactor building and ventilation systems are capable of minimizing the release of radioactivity to the atmosphere during potential abnormal conditions.

### Specification SR 5.5.1 - Reactor Building, Surveillance Requirements

The instrumentation which monitors the reactor building sub-atmospheric pressure will be functionally tested once every month and calibrated once a year.

### Basis for Specification SR 5.5.1

The reactor building atmosphere is normally maintained slightly below atmospheric pressure by the ventilation system (see FSAR Section 6.1.3.2). This requirement minimizes the amount and consequences of airborne activity released from the plant under most conditions (see FSAR Section 14.12.8). The leak rate of the building itself is not a significant parameter as is shown in FSAR Section 6.1.4.2.

Specification SR 5.5.2 - Reactor Building Pressure Relief  
Device, Surveillance

The reactor building overpressure relief system differential pressure switches shall be functionally tested on a monthly basis and calibrated annually.

The louver groups shall be individually exercised quarterly.

Quarterly louver testing may be performed while the reactor is in operation only if the following prerequisites are adhered to:

- a) Reactor shall be under normal steady state operating conditions.
- b) Primary coolant pressure is within the normal envelope for existing conditions.
- c) Reactor building ventilation system is operating per Technical Specifications.
- d) No radioactive gas waste releases are in progress, nor is fuel handling being performed.
- e) No airborne activity above background as indicated by the building activity monitors.
- f) Area radiation monitors and local alarms are operable per Technical Specifications.

- g) No surveillance testing is being performed on the reactor ventilation system or the radiation monitoring systems.
- h) Only one segment (group of louvers) of the louver system shall be tested at any given time.
- i) Communication shall exist between personnel performing the tests and the control room operators.
- j) Capability shall exist to manually shut the louver panels.
- k) Testing of the louver system shall not exceed a total duration of six (6) hours in any one quarter.
- l) Non-compliance with any of the above conditions will require testing to be discontinued and the louver system will be returned to normal.

The reactor building relief (louver) system shall be exercised annually.

Basis for Specification SR 5.5.2

The reactor building pressure relief device is designed to protect the building in the event that pressure in the Reactor Building exceeds the turbine building pressure by 3 inches of water. The device consists of louvers installed in a number of individual modules operated by mechanical linkages to pneumatic actuators (see FSAR

Section 6.1.3.4). The specified test frequency shall ensure the operability of the reactor building relief system.

Specification SR 5.5.3 - Reactor Building Exhaust Filters,  
Surveillance

The exhaust filters in the reactor building ventilation system shall be tested as follows:

- a) A laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shall be performed after each 4400 hours of operation of the unit, or following painting, fire or chemical\* release in any ventilation zone communicating with the unit. The results of laboratory carbon sample analysis from the unit shall show  $\geq 90\%$  radioactive methyl iodide removed when tested in accordance with ANSI N510-1975 (130°C, 95% R.H.).

\*Defined as any material which could reasonably be expected to interfere with the charcoal to adsorb methyl iodide.

- b) A halogenated hydrocarbon test shall be performed once per calendar year or after each replacement of a charcoal adsorber bank or after structural maintenance on the filter housing. Halogenated hydrocarbon

removal by the charcoal filters shall be  $\geq 99\%$  when conducted at normal flow conditions in accordance with the applicable portions of ANSI N510-1975.

- c) The HEPA filters shall be leak tested in place once per calendar year, after each complete or partial replacement of a HEPA filter bank, or after any structural maintenance on the filter housing, using cold DOP. Cold DOP removal by the HEPA filters shall be  $\geq 99\%$  when tested in accordance with the applicable portions of ANSI N510-1975.
- d) Flow distribution across the HEPA and charcoal filters will be tested with initial operation of the system and following any structural modification to the filter housings. Air distribution shall be demonstrated within  $\pm 20\%$  across the HEPA and charcoal filters when tested in accordance with ANSI N510-1975.
- e) Total pressure drop across the combined HEPA filter and charcoal adsorber banks shall be less than 6" H<sub>2</sub>O at filter design flow  $\pm 10\%$ .

#### Basis for Specification SR 5.5.3

The reactor building exhaust filter system is designed to filter the reactor building atmosphere prior to release to the facility vent stack during both normal and accident conditions of operation. The system consists of three 50%

capacity units, two of which are in continuous operation, with the third on standby.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to remove particulate matter from the air stream to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the atmosphere. Bypass leakage for the charcoal adsorbers, and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. The surveillance test frequencies specified establishes system performance capabilities. If system conditions are as specified, the calculated doses will be less than the guidelines stated in 10 CFR 100 for the accidents analyzed, as indicated in Sections 14.8 and 14.12 of the FSAR. Pressure drop across the combined HEPA filter and charcoal adsorber of less than 6 inches of water at the filter design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

The activated carbon adsorber in the affected unit should be replaced if a representative sample fails to pass the iodine removal efficiency test. Any HEPA filters found defective should be replaced.



If painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed, as required, for operational surveillance.

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5.6 EMERGENCY POWER SYSTEMS - SURVEILLANCE REQUIREMENTS

Applicability

Applies to the surveillance of the equipment supplying electrical power to the essential plant services.

Objective

To establish the minimum frequency and type of surveillance for equipment supplying electric power to the plant auxiliaries to ensure that the motive power sources required to safely shut down the plant is available.

Specification SR 5.6.1 - Standby Diesel Generator Surveillance

The surveillance of the standby diesel generators shall be as follows:

- a) Each standby generator set will be started and loaded to at least 50% of rated full load capacity once every week. The test shall continue for at least two hours to enable the engine(s) and the generator to attain their normal operating temperature.
- b) A loss of outside source of power and turbine trip shall be simulated twice annually with the interval between tests to be not less than four (4) nor greater than eight (8) months to demonstrate that the standby generators, automatic controls, and load sequencers are operable.

- c) The diesel engine protective functions shall be calibrated annually.
- d) The diesel engine exhaust temperature "shutdown" and "declutch" shall be functionally tested monthly and calibrated annually.

Basis for Specification SR 5.6.1

The weekly test of the standby diesel generator is to exercise the engine by operating at design temperature and to demonstrate operating capability. These tests will allow for detection of deterioration and failure of equipment.

Tests once a year during refueling will functionally test the standby generator system.

Specification SR 5.6.2 - Station Battery Surveillance

The surveillance of the station and PPS batteries shall be as follows:

- a) The specific gravity and voltage of the pilot cell and temperature of adjacent cells and overall battery voltage shall be measured every week.
- b) The specific gravity and voltage to the nearest 0.01 volt, temperature of every fifth cell and height of electrolyte shall be measured every three months.

- c) The station and PPS batteries shall be load-tested to partial discharge once a year during plant shutdown.

Basis for Specification SR 5.6.2

The type of battery surveillance called for in this specification has been demonstrated through experience to provide a reliable indication of a battery cell initial breakdown well before it becomes unserviceable. Since batteries will deteriorate with time, these periodic tests will avoid precipitous failure.

The manufacturer's recommendation for equalizing charge is vital to maintenance of the ampere-hour capacity of the battery. As a check upon the effectiveness of this charge, each battery will be load tested to determine its ampere-hour capacity. In addition, its voltage shall be monitored as a function of time. If a cell has deteriorated or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

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5.7 FUEL HANDLING AND STORAGE SYSTEMS - SURVEILLANCE  
REQUIREMENTS

Applicability

Applies to surveillance of the fuel handling and fuel storage systems during irradiated fuel handling and storage.

Objective

To ensure the prevention of any uncontrolled release of radioactivity during fuel handling and fuel storage by establishing the minimum frequency and type of surveillance on the equipment for the fuel handling and storage systems.

Specification      SR 5.7.1 - Fuel Handling Machine  
Surveillance

The surveillance of the fuel handling machine will be as follows:

- a) Prior to refueling, the fuel handling machine cooling water leak detector will be functionally tested.
- b) A functional test of the Fuel Handling Machine and Isolation Valve movements, interlocks, limit switches, and alarms will be performed or simulated prior to annual refueling periods.



Basis for Specification SR 5.7.1

The fuel handling machine provides for the safe refueling of the reactor. To assure the reliability of the fuel handling machine during the refueling operation, the machine and its associated interlocks, limit switches and alarms will be tested prior to refueling. All motions of the machine should be cycled, including the pick-up and release of a dummy element. A test of the helium system and the cooling system will be made. These checks will assure the capability to maintain the proper atmosphere environment within the machine to prevent any uncontrollable release of activity, proper purging and back filling capabilities, and the capability to maintain temperature of fuel elements within the machine below 750°F.

Specification SR 5.7.2 - Fuel Storage Facility  
Surveillance

The surveillance of the fuel storage facility will be as follows:

- a) The fuel storage facility helium pressure indicators and alarms will be calibrated and functionally tested annually.
- b) The fuel storage facility cooling system flow indicators, and flow and temperature alarms shall be calibrated and functionally tested annually.

Basis for Specification SR 5.7.2

The fuel storage wells are provided for safe storage of new and irradiated fuel elements. The basic design of the wells is to provide a low temperature dry helium environment. All conditions connected with this requirement are monitored by pressure, temperature, and flow sensitive devices. The temperature and flow detecting devices maintain surveillance of the wells' two independent cooling systems and are set to alarm at previously determined maximum or minimum values. The pressure sensitive device is available to guard against any over-pressurization of the wells. The specified annual surveillance interval is sufficient to insure proper operation of the instrumentation.

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5.8 RADIOACTIVE EFFLUENT DISPOSAL SYSTEMS - SURVEILLANCE  
REQUIREMENTS

Applicability

Applies to surveillance of the Radioactive Effluent Disposal Systems.

Objective

To establish the minimum frequency and type of surveillance on the equipment of the Radioactive Effluent Disposal Systems to assure that releases of radioactivity are within those specified in Section 4.8.

Specification SR 5.8.1 - Radioactive Gaseous Effluent  
System Surveillance

The surveillance of the radioactive gaseous waste disposal system shall be as follows:

- a) Automatic vent high activity blocking and transfer functions of the gaseous waste system shall be tested prior to each controlled release or once a month, whichever is more frequent.
- b) Automatic gaseous waste header high activity transfer to the gas waste vacuum tank shall be tested once per month.

- c) The gas waste header activity monitors shall be functionally tested once per month and calibrated quarterly.
- d) The vent monitor system shall be functionally tested weekly, calibrated quarterly, and following maintenance on the detector system.
- e) Flow recorders shall be calibrated annually.
- f) The vent iodine/particulate monitor filter shall be analyzed once per week.

Specification SR 5.8.2 - Radioactive Liquid Effluent  
System Surveillance

The surveillance of the radioactive liquid waste disposal system shall be as follows:

- a) The level alarms and pump interlocks on the two liquid waste receiver tanks and monitoring tank shall be tested once per year.
- b) The liquid effluent discharge blocking valve shall be functionally tested prior to each release or once a month, whichever is more frequent.
- c) The activity monitors of the liquid waste disposal line and the low cooling water blowdown flow switch shall be functionally tested prior to the controlled discharge of any liquid wastes or once a month,

whichever is more frequent. The activity monitors shall be calibrated quarterly and following maintenance on the detector system.

Basis for Specification SR 5.8.1 and SR 5.8.2

The frequency specified above is based upon industry experience and minimal disposal requirements of the plant. Tests prior to discharge using the installed check source mounted in the instrument will provide both a check on the calibration as well as a dynamic test of the various monitors, alarms, and protective functions.

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5.9 ENVIRONMENTAL SURVEILLANCE - SURVEILLANCE REQUIREMENTS

Applicability

Applies to sampling for environmental radioactivity in the vicinity of the plant.

Objective

To establish a sampling schedule which will recognize changes in radioactivity in the environs and assure that effluent releases are kept as low as practicable and within the limits of Appendix B, Table II, 10 CFR 20.

Specification      SR 5.9.1 - Environmental Radiation,  
Surveillance Requirements

a) Gaseous Release

Sampling of air, external gamma, milk, forage and crops shall be conducted in accordance with Action Guide 3 during the first three years of operation and thereafter in accordance with Table 5.9-1 and Table 5.9-2, as specified below:

- 1) If releases from the plant vent produced concentrations or exposures less than 3% of those specified in 10 CFR 20 for unrestricted areas and the general population during the previous quarter, the environmental survey shall be conducted in accordance with Action Guide 1 for the current quarter.

- 2) If the concentrations or exposures during the previous quarter were greater than 3% but less than 10% of those specified in 10 CFR 20 for unrestricted areas and the general population, the environmental survey shall be conducted in accordance with Action Guide 2 for the current quarter. If the samples taken under Action Guide 2 do not indicate any significant increase in environmental radioactivity, the survey shall revert to Action Guide 1.
- 3) If the concentrations or exposures during the previous quarter were greater than 10% of those specified in 10 CFR 20 for unrestricted areas and the general population, the environmental survey shall be conducted in accordance with Action Guide 3 for the current quarter. If the samples taken under Action Guide 3 do not indicate any significant increase in environmental radioactivity, the survey shall revert to Action Guide 2.

b) Liquid Release

Sampling of water and silt, potable water, and aquatic biota shall be conducted in accordance with Action Guide 3 during the first three years of operation and thereafter in accordance with Table 5.9-1 and Table 5.9-2 as specified below:

- 1) If the gross beta-gamma activity released from the station during the previous quarter was less than 3% of  $MPC_w$ , the environmental survey shall be conducted in accordance with Action Guide 1 for the current quarter.
- 2) If the gross beta-gamma activity released from the station during the previous quarter was greater than 3%  $MPC_w$  but less than 10%  $MPC_w$ , the environmental survey shall be conducted in accordance with Action Guide 2 for the current quarter. If the samples taken under Action Guide 2 do not indicate any significant increase in environmental radioactivity, the survey shall revert to Action Guide 1.
- 3) If the gross beta-gamma activity released from the station during the previous quarter was greater than 10% of  $MPC_w$ , the environmental survey shall be conducted in accordance with Action Guide 3 for the current quarter. If samples taken under

Action Guide 3 do not indicate any significant increase in environmental radioactivity, the survey shall revert to Action Guide 2.

- 4) Results of the aquatic biota sampling program will be reviewed with appropriate agencies after one year of sampling following commercial operation to establish the required extent of future sampling.

Basis for Specification SR 5.9.1

Programs for monitoring the environment in the vicinity of Ft. St. Vrain will be conducted by Colorado State University under a contract from Public Service Company of Colorado (the licensee) and by the Colorado Department of Health with assistance by the Environmental Protection Agency's Western Environmental Radiation Laboratory. The Colorado Department of Health program includes sampling and analyses of air, water and milk. In addition, they will have special programs for sampling tritium in surface water and atmospheric concentrations of  $^{85}\text{Kr}$ .

A preoperational radiological monitoring program has been conducted since March 1969. This program has established an adequate baseline to which operational environmental data can be compared.

The operational environmental surveillance program will be maintained on a continuous basis to verify that projected and anticipated concentrations of radioactive materials in

the environment are not exceeded. The extent to which environmental monitoring programs are conducted should depend on the actual release of radioactivity into the environment.

When the quantity of material released is small the environmental monitoring program may be minimal. For larger releases of radioactive material, a more comprehensive environmental monitoring program is appropriate. The surveillance levels specified in Action Guide 1 and Action Guide 2 are comparable to intake Range 1 and Range 2 as given in Federal Radiation Council Report No. 2.

The operational surveillance program provides for collection and analyses for samples within an area extending to a twenty mile radius from the reactor. A concentrated area of sampling within a one mile radius is designated the facility zone; the area from one to ten miles is called the adjacent zone, and the reference zone is from ten to twenty miles.

Table 5.9-2 gives the location of each sampling station and the types of samples to be taken at each station. Table 5.9-3 gives the minimum sensitivities for the various analyses and/or measurements made on the samples. Figure 5.9-1 and Figure 5.9-2 indicate the sample station locations.

The aquatic biota sampling program is a supplemental part of the Environmental Surveillance Monitoring Program and was not a factor in the design of the basic sampling program which was designed on the basis of critical pathways to man.

It is felt that sampling during the preoperational phase and for a representative period following operation will adequately demonstrate any potential effect of plant operation on aquatic biota. Therefore, it is planned that the results of the aquatic biota sampling program will be reviewed with representatives from interested agencies such as the Bureau of Sport Fisheries and Wildlife and the Colorado Game, Fish and Parks Department following one year of commercial operation to establish the extent to which sampling should be continued beyond that point.

For further information, see FSAR Section 2.7.

TABLE 5.9-1

ENVIRONMENTAL RADIATION SURVEILLANCE PROGRAM SCHEDULE

EXPOSURE ROUTES OR MEDIA & SAMPLE TYPES (NUMBER OF LOCATIONS)	SAMPLING FREQUENCIES AND ANALYSES BY ACTION LEVELS (BASED UPON ACTUAL EMISSIONS AS PERCENTAGES OF RELEASE RATES AUTHORIZED BY 10 CFR 20)		
	Action Level 1: Less Than 3%	Action Level 2: 3% to 10%	Action Level 3: Greater Than 10%
EXTERNAL EXPOSURE	Average mR/day determined by	Average mR/day determined by	Average mR/day determined by
TLD Chips (36 Locations)	QUARTERLY cumulative exposures; collection and analysis in rotation of 1/3 of all TLD's MONTHLY.	QUARTERLY cumulative exposures; collection and analysis in rotation of 1/3 of all TLD's MONTHLY.	MONTHLY analysis of all TLD's.
ATMOSPHERE	Gross beta, every filter, WEEKLY; gamma spectrum of filter and cartridge composites, MONTHLY.	Same as for Level 1, plus gross alpha on one weekly set of filters, MONTHLY.	Gross alpha and beta, every filter; gamma spectrum of filter and cartridge composites, all WEEKLY.
Membrane Filters for Particulates; Charcoal Cartridges for Iodine. (7 Locations)			
Tritium Oxide (2 Locations: F1 and F4)	Specific activity of tritium in atmospheric water vapor by passive absorption and liquid scintillation counting, QUARTERLY.	Specific activity of tritium in atmospheric water vapor by passive absorption and liquid scintillation counting, MONTHLY.	Specific activity of tritium in atmospheric water vapor by passive absorption and liquid scintillation counting, WEEKLY.
WATER	Gross beta, tritium, and gamma spectrum analyses; facility area and nearest off-site supply (shallow wells at town of Gilcrest, 6 miles northeast), QUARTERLY.	Gross beta, tritium, and gamma spectrum analyses; facility area and nearest off-site supply (shallow wells at town of Gilcrest, 6 miles northeast), MONTHLY.	Gross beta, tritium, and gamma spectrum analyses; facility area and nearest off-site supply (shallow wells at town of Gilcrest, 6 miles northeast), MONTHLY, plus Sr 89 and 90 analyses.
Potable Water (2 Locations)			



TABLE 5.9-1

ENVIRONMENTAL RADIATION SURVEILLANCE PROGRAM SCHEDULE (Cont'd)

EXPOSURE ROUTES OR MEDIA & SAMPLE TYPES (NUMBER OF LOCATIONS)	SAMPLING FREQUENCIES AND ANALYSES BY ACTION LEVELS (BASED UPON ACTUAL EMISSIONS AS PERCENTAGES OF RELEASE RATES AUTHORIZED BY 10 CFR 20)		
	Action Level 1: Less Than 3%	Action Level 2: 3% to 10%	Action Level 3: Greater Than 10%
Precipitation (2 Locations: F1 and F4)	No collection or analyses of precipitation at Level 1.	Gross beta, MONTHLY.	Gross beta, tritium, and Sr 89 and 90, MONTHLY; gamma spectrum of composite, QUARTERLY.
Surface Water and Silt (7 Locations)	Gross beta, tritium, and gamma spectrum, QUARTERLY.	Same as for Level 1, but MONTHLY.	Same as for Level 2, plus Sr 89 and 90 analyses, MONTHLY.
<u>FOOD CHAINS</u>			
Soil, Forage, and Crops (13 Locations)	Tritium and gamma spectrum analyses of forage and crops in the most probable routes to man. QUARTERLY, as available (i.e., spring, summer, and fall).	Tritium and gamma spectrum analyses of forage and crops in the most probable routes to man. MONTHLY during growing season (i.e., approximately April to October).	Tritium and gamma spectrum analyses of forage and crops in the most probable routes to man. Same as Level 2, plus Sr 89 and 90, plus concurrent soil samples analyzed for the same nuclides. MONTHLY during growing season.
Beef Cattle (1 Location: Facility Area)	No analysis of beef at Level 1.	Gamma spectrum, tritium, and Sr 89 and 90 analyses on one meat sample from beef raised in facility area; ANNUALLY, at end of grazing season (i.e., late fall).	Same as for Level 2, plus total body count of 2 to 4 animals from facility area, QUARTERLY.

TABLE 5.9-1

ENVIRONMENTAL RADIATION SURVEILLANCE PROGRAM SCHEDULE (Cont'd)

EXPOSURE ROUTES OR MEDIA & SAMPLE TYPES (NUMBER OF LOCATIONS)	SAMPLING FREQUENCIES AND ANALYSES BY ACTION LEVELS (BASED UPON ACTUAL EMISSIONS AS PERCENTAGES OF RELEASE RATES AUTHORIZED BY 10 CFR 20)		
	Action Level 1: Less Than 3%	Action Level 2: 3% to 10%	Action Level 3: Greater Than 10%
Milk (13 Locations)	Tritium, gamma spectrum, and Sr 89 and 90 analyses on composite. Facility area only, QUARTERLY.	Tritium, gamma spectrum, and Sr 89 and 90 analyses on composite. Facility, adjacent and reference areas; MONTHLY during pasture season, otherwise QUARTERLY.	Tritium, gamma spectrum, and Sr 89 and 90 analyses on composite. Same as for Level 2, but WEEKLY during pasture season, otherwise MONTHLY.
AQUATIC BIOTA			
2 Streams, Above and Below Discharge Point	Gross beta and gamma spectrum analyses of composites of each of 4 categories: 1) suspended organisms, 2) benthic organisms, 3) vascular plants, and 4) fish. QUARTERLY, as available.	Gross beta and gamma spectrum analyses of composites of each of 4 categories: 1) suspended organisms, 2) benthic organisms, 3) vascular plants, and 4) fish. MONTHLY during summer; otherwise QUARTERLY, as available.	Gross beta and gamma spectrum analyses of composites of each of 4 categories: 1) suspended organisms, 2) benthic organisms, 3) vascular plants, and 4) fish. Same as for Level 2, plus Sr 89 and 90 analyses.

TABLE 5.9-2

SAMPLE TYPES AND LOCATIONS FOR ENVIRONMENTAL SURVEILLANCE IN THE  
VICINITY OF THE FORT ST. VRAIN NUCLEAR GENERATING STATION  
(Revised April 1971)

LOC.	Media Sampled At Location						Compass Direc-	Description of Sampling Location <sup>c</sup>
	I.D.	Air	M&F	TLD	H <sub>2</sub> O	SED	AQB <sup>b</sup>	
F-1	*			*				N Potato cellar at W end of farm site just across creek N of reactor; TLD on pole at NE corner.
F-2	*			*				E A cabin .3 mile E and .4 mile N of site #1.
F-3	*			*				SE Old dairy barn on Robert Bruce farm 1/2 mile SE of reactor; TLD on first pole N of driveway.
F-4	*			*				S Small shed behind the Maul farmhouse 1/2 mile SW of reactor; TLD on pole W of house.
F-7				*				NE .3 mile E of #1; on the pole by gate where road turns S.
F-8				*				NE .3 mile S of #7; on second pole S of gate on hill.
F-9				*				SE .2 mile S and .3 mile W of #3; on 7th pole W of corner.
F-10				*				S On 8th pole W of #9; directly across road from pump house.
F-11				*				S On 3rd pole W of #10.
F-12				*				SW On 4th pole N of #4.
F-13				*				SW On 5th pole N of #12.
F-14				*				NW 5.5 miles S and .9 mile E of Johnstown. On pole at corner in road.
F-44	*			*				N Ted Horst farm, due N of reactor site.

b - With the exception of the winter months.

c - Sampling location description provided for information only. Locations shall be changed as conditions warrant.

TABLE 5.9-2

SAMPLE TYPES AND LOCATIONS FOR ENVIRONMENTAL SURVEILLANCE IN THE  
VICINITY OF THE FORT ST. VRAIN NUCLEAR GENERATING STATION  
(Revised April, 1971) (Cont'd)

LOC.	Media Sampled At Location						Compass Direc- tion	Description of Sampling Location <sup>c</sup>
	I.D.	Air	M&F	TLD	H <sub>2</sub> O	SED	AQB <sup>b</sup>	
A-5	*	*	*					NE Air sampler at Lloyd Rumsey farm. TLD on 1st pole N of Ernest Wolfrum farm. Milk and forage at Floyd Nicols farm.
A-6	*	*	*					SE Calving shed on the Clifton Wissler dairy farm. TLD on 1st pole E of driveway.
A-15		*	*					NW Paul Bader dairy, 4.2 miles W of Hwy 87 on Hwy 60; TLD on pole on W side of driveway.
A-18			*					N 3 miles S and 1.5 miles W of #19, on SE pole of intersection, across street from old school.
A-27			*					NW 1 mile E of Hwy 87 on Hwy 56, then 1 mile S on 1st pole on NE side of corner.
A-28		*	*					NW Norman Carlson dairy, 2 miles E of Hwy 87 on Hwy 60 and 1 mile S. TLD on pole directly W of driveway.
A-29			*					N 3 miles S and 1.6 miles E of Johnstown, on E pole on S side of road.
A-30			*					NE 1 mile S on Hwy 256 on Hwy 60 on middle pole at NE side of intersection.
A-31		*	*					E Oliver Loren dairy, 1.5 miles S of Peckham; TLD on 1st pole S of driveway.
A-32			*					E 2.5 miles S and 2 miles W of #31, on NW pole at intersection.

b - With the exception of the winter months.

c - Sampling location description provided for information only. Locations shall be changed as conditions warrant.

TABLE 5.9-2

SAMPLE TYPES AND LOCATIONS FOR ENVIRONMENTAL SURVEILLANCE IN THE  
VICINITY OF THE FORT ST. VRAIN NUCLEAR GENERATING STATION  
(Revised April, 1971) (Cont'd)

LOC.	Media Sampled At Location						Compass Direc- tion	Description of Sampling Location <sup>c</sup>
	I.D.	Air	M&F	TLD	H <sub>2</sub> O	SED	AQB <sup>b</sup>	
A-33			*	*				E Miller Brothers dairy, .2 mile S of junction of Hwy's 66 and 85 and .5 mile E; TLD on 1st pole W of driveway.
A-34				*				S 1 mile E of junction of Hwy's 87 and 254 on SW pole at intersection.
A-35				*				SW 2 miles E of Hwy 87 on Hwy 66 on SW pole at intersection.
A-36			*	*				W Milk and forage at Marvin Coonts farm. TLD at Phillip Ball dairy.
R-16				*				NW Going S of Hwy 87, on pole beside yield sign on Loveland off-ramp.
R-17			*	*				N Ed Borensen dairy, 8 miles E of Hwy 87 on Hwy 34, then .3 mile S; TLD on pole in front of trailer house on W side of road, i.e., 10 poles from Hwy 34.
R-19				*				N 4 miles E of #17 on Hwy 34, on pole at SW corner of intersection (golf course across street from pole).
R-20			*	*				NE Wally Kaufman dairy, .5 mile E of LaSalle and 1.6 miles S; TLD on pole W of dairy barn on W side of road.
R-21				*				E 2.5 miles S and 2 miles E of #20, on SW pole at intersection.
R-22			*	*				SE Hagan Brothers Dairy, 4.3 miles E of Ione; TLD on 1st pole E of driveway.

b - With the exception of the winter months.

c - Sampling location description provided for information only. Locations shall be changed as conditions warrant.



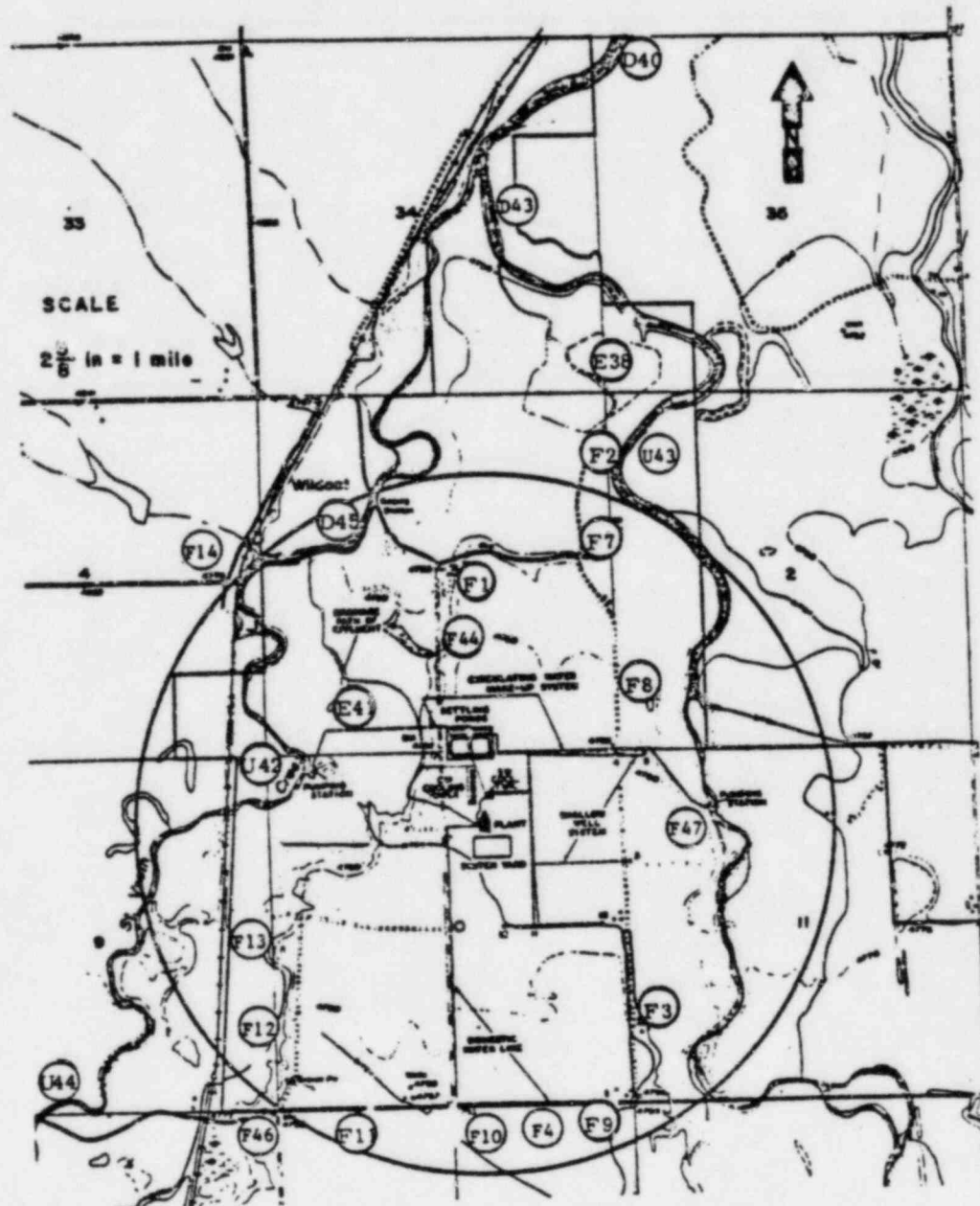
TABLE 5.9-2

SAMPLE TYPES AND LOCATIONS FOR ENVIRONMENTAL SURVEILLANCE IN THE  
VICINITY OF THE FORT ST. VRAIN NUCLEAR GENERATING STATION  
 (Revised April, 1971) (Cont'd)

LOC.	Media Sampled At Location						Compass Direc-	Description of Sampling Location <sup>c</sup>
	I.D.	Air	M&F	TLD	H <sub>2</sub> O	SED	AQB <sup>b</sup>	
k-23		*	*					S Ernest Terry dairy, 2.3 miles W of Ft. Lupton on Hwy 52; TLD on 1st pole W of driveway.
R-24				*				SW Going S on Hwy 87, on pole by stop sign on Frederick/Dacono off-ramp.
R-25		*	*					SW Milk and forage at Angelo Bendegna farm on Oxford Road. TLD near Paul Knutson farm.
R-26				*				W 4.7 miles N of Hwy 66 on Hwy 287 on SE pole at intersection.
E-38					*	*	*	NE Goosequill Pond, .3 mile E and .4 mile N of site #1.
E-41				*	*			W Slough. Due W of reactor.
U-42				*	*	*		SW St. Vrain from bridge; .7 mile W of site #4.
U-43				*	*	*		NE South Platte, 1 mile upstream from Goosequill discharge.
U-42							*	E St. Vrain at pumping station due E of reactor.
D-45				*	*	*		NW St. Vrain Creek S of bridge near gauging station.
D-39				*				NE In town of Gilcrest, Co-op store tap water.
D-40				*	*	*		N 1/2 mile below junction of the St. Vrain and Platte.
D-37				*	*			NE Lower Latham Reservoir; 2.5 miles E of LaSalle.
D-43							*	N South Platte 1/2 mile downstream from Goosequill discharge.

b - With the exception of the winter months.

c - Sampling location description provided for information only. Locations shall be changed as conditions warrant.



**Figure 5.9-1** Sampling stations within the facility area (on site)  
 Revised 5-6-71.



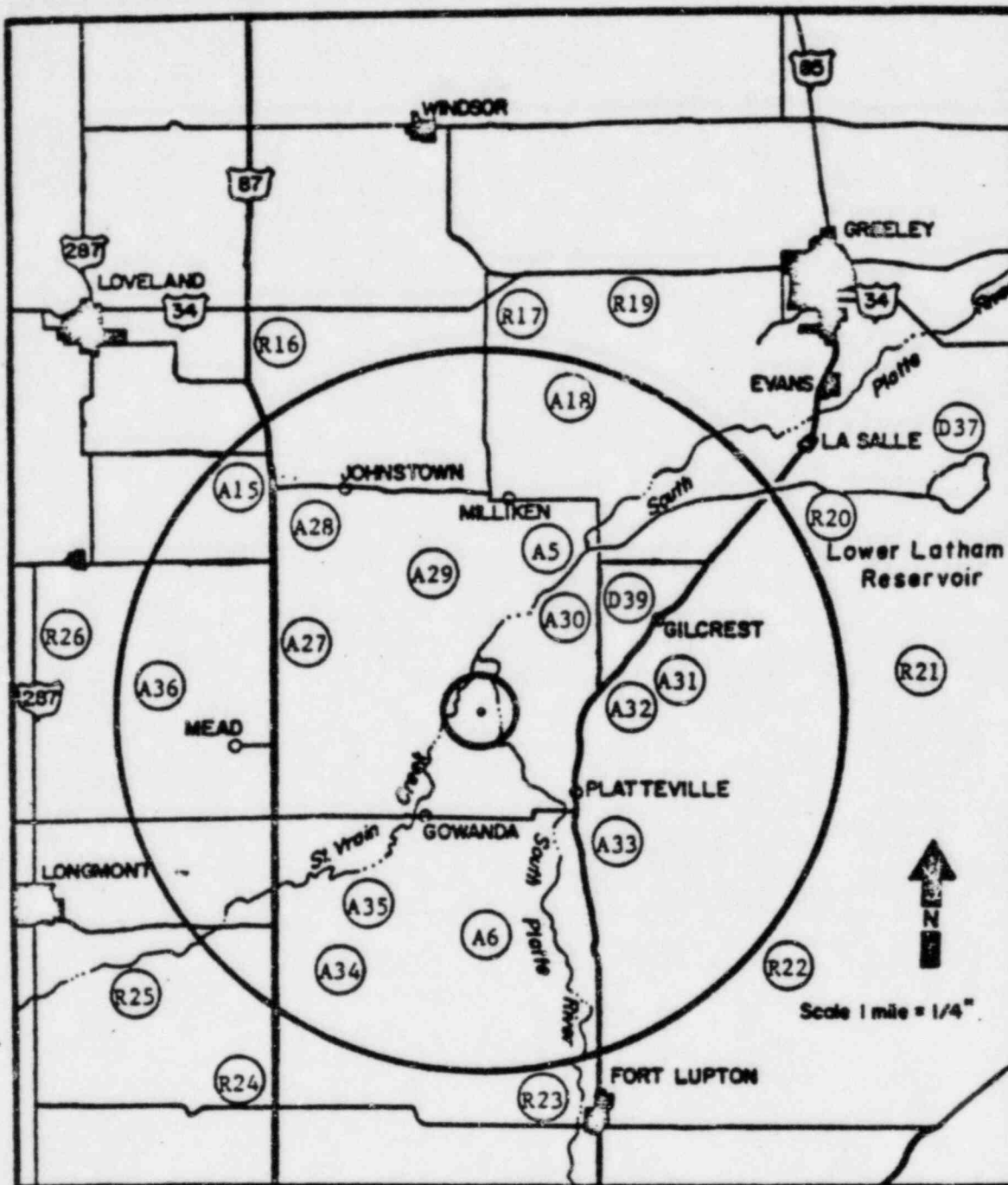


Figure 5.9-2 Sampling stations in the adjacent and reference areas (off site). Revised 5-6-71.

TABLE 5.9-3

SENSITIVITIES FOR ENVIRONMENTAL RADIATION MEASUREMENTS

Media	Isotope	Counting Efficiency	Analytical Technique	M.D.A. 99% C.L.
Air	Gross $\alpha$	48%	Int. Prop. Ctr.	(a)
	Gross $\beta$	26%	Low Beta G. M.	0.01 pCi/m <sup>3</sup>
	<sup>137</sup> Cs	21%	Gamma Spect.	0.03 pCi/m <sup>3</sup>
	<sup>95</sup> Zr- <sup>95</sup> Nb	30%	Gamma Spect.	0.02 pCi/m <sup>3</sup>
	<sup>106</sup> Ru	4.3%	Gamma Spect.	0.20 pCi/m <sup>3</sup>
	<sup>144</sup> Ce	6.8%	Gamma Spect.	0.10 pCi/m <sup>3</sup>
	<sup>131</sup> I	(a)	Gamma Spect.	(a)
	<sup>3</sup> H	25%	Liquid Scint.	6.8 pCi/m <sup>3</sup> (d)
External Gamma			Thermoluminescent Read-out	40 mR (b)
Forage	Gross $\beta$	8%	Low Beta G.M.	3.0 pCi/g
	<sup>137</sup> Cs	(a)	Gamma Spect.	35 pCi
	<sup>3</sup> H	25%	Liquid Scint.	2000 pCi/l
Milk	<sup>137</sup> Cs	6.7%	Gamma Spect.	2.0 pCi/l
	<sup>131</sup> I	(a)	Gamma Spect.	(a)
	<sup>89</sup> , <sup>90</sup> Sr	8%	Chem. Separation	10.0 pCi/l
	<sup>3</sup> H	25%	Liquid Scint.	2000 pCi/l

TABLE 5.9-3

SENSITIVITIES FOR ENVIRONMENTAL RADIATION MEASUREMENTS (Cont'd)

Media	Isotope	Counting Efficiency	Analytical Technique	M.D.A. 99% C.L.
Water	$^{137}\text{Cs}$	6.7%	Gamma Spect.	2.0 pCi/l
	Gross $\beta$	(c)	Low Beta G.M.	1.0 pCi/l
	$^3\text{H}$	25%	Liquid Scint.	2000 pCi/l
Sediment	Gross $\beta$	8%	Low Beta G.M.	3.0 pCi/g

- (a) To be determined.
- (b) Radiation exposure necessary to produce a response equal to 3 sigma of the background as determined over a three month period.
- (c) Dependent upon amount of dissolved and suspended solids.
- (d) Depends upon relative humidity.

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## 5.10 FIRE SUPPRESSION SYSTEMS - SURVEILLANCE REQUIREMENTS

### Applicability

Applies to the surveillance of the fire suppression and protection systems and equipment.

### Objective

To establish the minimum frequency and type of surveillance on the equipment of the fire suppression and protection equipment to assure that the capability exists for suppressing any fire involving safety related equipment.

### Specification SR 5.10.1 - Three Room Control Complex HVAC System, Surveillance Requirement

The HVAC isolation dampers and associated fans of the control room, auxiliary electric room, and the 480 volt switchgear room, shall be tested annually to verify correct response to a simulated actuation signal from the Halon Fire Suppression System.

### Basis for Specification SR 5.10.1

Annual testing of the room isolation dampers and associated fans of the three-room control complex is sufficient to demonstrate capability to operate when required by the Halon Fire Suppression System. The dampers in the ventilation systems of the control and auxiliary electric rooms, and of the 480 volt switchgear

room automatically close upon actuation of the Halon Fire Suppression System for that room. At the same time, the various ventilation fans associated with these areas are also tripped off or prevented from starting. The isolation of the ventilation system concurrent with Halon discharge is required to maintain effective concentrations of Halon in the area of the fire.

Specification SR 5.10.2 - Halon Fire Suppression System,  
Surveillance Requirements

Operability of the Halon fire suppression system for the control room, auxiliary electric equipment room, and 480 volt switchgear room shall be demonstrated as follows:

- a) Quarterly, verify that Halon Storage Cylinder weight is at least 95% of full rated charge.
- b) Quarterly, verify that Halon Storage Cylinder pressure is at least 90% of full rated charge.
- c) Annually, verify that distribution headers and nozzles are open by flowing clean air at low pressure through the system.
- d) Annually, verify response of system to an actuation signal by disconnecting each solenoid coil and measuring the voltage created by the actuating signal.

Basis for Specification SR 5.10.2

Quarterly, check-weighing and pressure verification of the Halon Fire Suppression System cylinders meets the requirements of NFPA Code Section 12A for fire suppression system operability tests. Annual verification that the distribution headers are not plugged demonstrates their ability to distribute Halon when needed to suppress a fire, while measurement of an actuation signal to the solenoid coil of each cylinder release mechanism gives extra assurance that the system will be capable of performing its design function when required.

Specification SR 5.10.3 - Smoke Detectors and Alarm, Surveillance Requirement

The smoke detectors and alarms listed in Table 4.10-3 shall be demonstrated operable as follows:

- a) Monthly by functional test of the non-supervised circuits between the local panels and the main control panel.
- b) Semi-annually by functional test of the smoke detectors and alarms, and by functional test of supervisory circuits associated with the smoke detector alarms.



Basis for Specification SR 5.10.3

Operability of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Testing at the specified intervals is sufficient to ensure operability of the system.

Specification SR 5.10.4 - Fire Barrier Penetration Seal, Surveillance Requirements

Fire Barrier Penetration Seals shall be visually inspected:

- a) Annually, to verify each remains intact, or
- b) Immediately following any maintenance which disturbs the retardant material, to verify seal is returned to its previous condition.

Basis for Specification SR 5.10.4

Inspection of penetration seals either annually or following maintenance assures that the Fire Barrier Penetration Seals remain unchanged.

Specification SR 5.10.5 - Breathing Air System,  
Surveillance Requirement

The operability of the Breathing Air System shall be demonstrated annually, as follows:

- a) Functionally test the compressors and air supply piping.
- b) Test the quality of the air supplied.

Basis for Specification SR 5.10.5

The Breathing Air System and associated piping supplies compressed air for recharging air cylinders of self-contained breathing apparatus and to the control room for personnel use with approved breathing equipment, in a Halon or toxic gas atmosphere. By functionally testing the system annually, assurance of breathing air availability and quality is provided.

Specification SR 5.10.6, Fixed Water Spray System,  
Surveillance Requirement

Each of the fixed water spray systems listed in LCO 4.10.5 shall be verified operable as follows:

- a) Annually by cycling each valve in the flow path through one complete cycle of full travel.

- b) Each refueling cycle by inspection of the flow headers to verify their integrity and inspection of each nozzle to verify no blockage.
- c) At least once per three years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.
- d) The temperature instruments and controls associated with the reactor plant exhaust filters shall be functionally tested semi-annually.

Basis for Specification SR 5.10.6

Operation of the valves and verification of the flow path at the specified intervals is sufficient to demonstrate capability to operate if required.

Semi-annually testing the temperature instruments and controls associated with the reactor plant exhaust filters is sufficient to ensure operability of the system.

Specification SR 5.10.7 - Carbon Dioxide Fire Suppression System, Surveillance Requirement

The carbon dioxide fire suppression system shall be demonstrated operable as follows:

- a) Weekly - verify storage tank level and pressure.

b) Annually:

- 1) Verify operation of system valves and associated dampers upon actuation signal.
- 2) Verify flow from each nozzle during a "puff test."

Basis for Specification SR 5.10.7

A weekly check of level and pressure in the carbon dioxide storage tank insures sufficient carbon dioxide for fire suppression and the support equipment is operating properly.

An annual flow check and simulated automatic actuation of the system along with the regular calibration of the system instrumentation provides adequate assurance that the system will be ready to suppress any fire that could occur in the emergency diesel generator rooms.

Specification SR 5.10.8 - Fire Hose Stations, Surveillance Requirement

Each of the fire hose stations listed in Table 4.10-7 shall be checked monthly to insure all required equipment is at the station.

The fire hoses at these stations shall be removed for inspection, repacking, and refurbishing as required once per refueling cycle.

These fire hose stations shall be tested for flow and the fire hoses hydrostatically tested once every 3 years.

Basis for Specification SR 5.10.8

These checks of the fire water hose system will demonstrate the system's ability to operate if required.

Specification SR 5.10.9 - Yard Fire Hydrants and Hydrant Hose Houses, Surveillance Requirement

Each of the yard fire hydrants and associated hydrant hose houses listed in LCO 4.10.8 shall be verified operable as follows:

- a) Monthly by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.
- b) Semi-annually (once during March, April, or May, and once during September, October, or November) by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.
- c) Annually by conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any yard fire hydrant and by replacement of all degraded gaskets in couplings.

Basis for Specification SR 5.10.9

Inspection and testing at the specified intervals is sufficient to ensure operability of the hydrants and hydrant hose houses when required.

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ATTACHMENT 4

SAFETY ANALYSIS REPORT FOR THE FORT ST. VRAIN IN-SERVICE  
INSPECTION AND TESTING PROGRAM UPDATE

SAFETY ANALYSIS REPORT FOR THE FSV IN-SERVICE INSPECTION AND  
TESTING PROGRAM UPDATE

1. BACKGROUND

The Fort St. Vrain in-service inspection and testing program is specified by the Plant Technical Specification Surveillance Requirements (Ref. 1).

In response to a commitment in the 1972 Safety Evaluation Report (Ref. 2) Public Service Company has been reviewing, as a continuing effort, the in-service inspection and testing program for Fort St. Vrain to feedback the acquired operating experience with the plant, and to update the program in light of more recent rules and regulations.

The original 1972 Safety Evaluation Report (Ref. 2) includes a commitment to review the in-service inspection program for the primary coolant system after five years of reactor operation.

The status of the review effort was originally described by Public Service Company, together with the planned approach to follow in conforming with the 1972 Safety Evaluation Report commitment (Ref. 3). A review of Public Service Company plans was performed by the Nuclear Regulatory Commission, who also identified priority items to be addressed beyond the scope of the original Safety Evaluation Report commitment (Ref. 4). The general in-service inspection and testing program review plan and the priority items were further discussed in letters and at meetings between the Nuclear Regulatory Commission and Public Service Company until a basic agreement was reached between both parties (Ref. 5 through 10). A schedule was established for the review of surveillance requirements for all major plant systems and equipment by subdividing them in priority categories as requested by the Nuclear Regulatory Commission (Ref. 11).

This Safety Analysis Report addresses the changes to Technical Specification Surveillance Requirements proposed by Public Service Company as a result of priority category I reviews.

2. METHODOLOGY FOR REVIEW OF ISI PROGRAM

2.1 Rules

A set of general rules governing the Fort St. Vrain in-service inspection and testing program review was established by Public Service Company in agreement with the Nuclear Regulatory Commission, as discussed above. Specific important features of these rules were that Section XI Division 2 of the ASME Code, then in form of a draft, could not be applied directly to Fort St. Vrain but was to be used as a guide; Public Service Company would provide the Nuclear Regulatory Commission with a comparison of recommended surveillance requirements with these proposed Code

requirements and justify the differences. It was also agreed that Public Service Company would rank the equipment by their importance to plant safety and specify surveillance requirements commensurate with that importance to safety. Detailed definitions and criteria for examinations and tests were developed accordingly and transmitted to the Nuclear Regulatory Commission (enclosure 3 to Ref. 12).

## 2.2 Review Process

### 2.2.1 PSC Preliminary Submittals

The results of all priority Category I reviews were transmitted to the Nuclear Regulatory Commission (Ref. 12, 13, 14). These reviews covered the major systems and components that are important to safety, including the prestressed concrete reactor vessel, the reactor internals, the reactor primary coolant system, the reactor secondary coolant system, and the PCRV auxiliary system. Each of the above packages contained, for Nuclear Regulatory Commission review, draft modifications to the Fort St. Vrain Technical Specification Surveillance Requirements along with an evaluation of the existing and proposed inspections and tests, an identification of ASME Code Section XI requirements, and a discussion of the differences.

### 2.2.2 Independent Review of PSC Preliminary Submittals

At the request of the Nuclear Regulatory Commission, the Public Service Company preliminary submittals were independently reviewed under the direction of the Los Alamos National Laboratory (Ref. 15, 16, 17). A first meeting was held at the Fort St. Vrain plant on November 20, 1981, between Los Alamos National Laboratory, Public Service Company, and their consultants (Ref. 18). The result of the independent review were subsequently included in a report, which was transmitted to both the Nuclear Regulatory Commission and Public Service Company (Ref. 19). Most of Public Service Company's recommended modifications to the Technical Specification Surveillance Requirements were found to be adequate and acceptable, and a few additional investigations were recommended. The recommendations resulting from the independent review were subsequently analyzed by Public Service Company who prepared a response to agree with or clarify these recommendations, or further support Public Service Company's position (Ref. 20). The responses were also reviewed independently, and the remaining open items were resolved at a final meeting held on July 29, 1982, at the Fort St. Vrain plant between the Nuclear Regulatory Commission, Los Alamos National Laboratory, Public Service

Company, and their consultants (Ref. 21, 22). At this meeting, it was agreed that Public Service Company would submit final proposed changes to the Technical Specification Surveillance Requirements, together with an implementation schedule.

#### 2.2.3 Additional PSC Preliminary Submittals

As a result of the independent review, Public Service Company has prepared additional preliminary submittals to the Nuclear Regulatory Commission (Ref. 23) which include draft modified Technical Specification Surveillance Requirements, as well as an evaluation of the additional examinations and tests.

#### 2.2.4 Final Proposed Changes to Technical Specification Surveillance Requirements

The final proposed changes to Technical Specification Surveillance Requirements were prepared by Public Service Company to reflect the draft proposed changes included in the preliminary submittals, modified as appropriate to take into account the results of the independent review, and the changes included in the additional preliminary submittals.

### 3. EVALUATION AND CONCLUSIONS

The proposed changes to the Technical Specification Surveillance Requirements generally expand the scope of in-service examination and testing that is currently performed at the Fort St. Vrain Nuclear Generating Station. This, in essence, provides greater assurance of plant safety and reliability.

Individual surveillance requirements have been evaluated in detail in Public Service Company preliminary submittals covering the PCRV, the primary coolant systems, and the secondary coolant system, and these evaluations are incorporated herewith by reference. The results of these reviews revealed that existing surveillance requirements were generally adequate in light of the newly defined methodology, which accounts for plant operating experience, importance to safety, unique design features and limitations, and ASME Code development for large HTGR designs. Minor modifications to surveillance intervals were made to reflect operating experience, and to provide operating flexibility. Additional tests were included to assure the operability and accuracy of instrumentation which can be used for monitoring the structural integrity of major plant equipment. Additional component testing was recommended, as a result of detailed reviews of plant systems, either when components important to safe plant shutdown and cooling were not in the scope of the current Technical Specifications, or when the testing method could be improved to provide additional assurance of component reliability. Additional component examinations were

also recommended, where appropriate, to provide further assurance concerning the continued structural integrity of critical welded joints, bolted connections, and load bearing features.

Through the independent review process, it was verified that: (1) The Public Service Company evaluations were technically correct; (2) uncovered deficiencies were resolved to the Nuclear Regulatory Commission's satisfaction; (3) sufficient examination and testing is planned, using the state of the art within limitations inherent to the Fort St. Vrain design, to assure the long term safety of the plant.

Since the proposed changes to the Technical Specification do not result from modifications to plant equipment, nor do they involve reductions in the margins of safety, it can be concluded that no unreviewed safety question is raised. Besides the proposed changes, no changes are required to other parts of the Technical Specifications.

#### 4. REFERENCES

1. Plant Technical Specifications
2. Safety Evaluation Report of January 20, 1972, Section 3.3.
3. Public Service Company letter dated October 13, 1978 (P-78169), In-service Inspection - Fort St. Vrain.
4. Nuclear Regulatory Commission letter dated January 15, 1979, In-service Inspection and Testing Program for Fort St. Vrain.
5. Public Service Company letter dated March 15, 1979 (P-79058), In-service Inspection Program for Fort St. Vrain.
6. Nuclear Regulatory Commission letter dated June 5, 1979, Summary of Meeting Held on May 2, 1979, to Discuss In-service Inspection.
7. Public Service Company Progress Report. Meeting held on August 20, 1979, between the Nuclear Regulatory Commission and Public Service Company.
8. Public Service Company letter dated August 22, 1979 (P-79176), Fort St. Vrain In-service Inspection and Testing Program.
9. Nuclear Regulatory Commission letter dated October 5, 1979, Proposed Plan of In-service Inspection and Testing for Fort St. Vrain.
10. Public Service Company Progress Report. Meeting held on November 1, 1979, between the Nuclear Regulatory Commission and Public Service Company.



11. Public Service Company letter dated November 30, 1979 (P-79289), Fort St. Vrain In-service Inspection and Testing Program.
12. Public Service Company letter dated February 8, 1980 (P-80014), Fort St. Vrain In-service Inspection and Testing - PCRV Auxiliary System.
13. Public Service Company letter dated March 3, 1980 (P-80034), Fort St. Vrain In-service Inspection and Testing (PCRV and PCRV Internals).
14. Public Service Company letter dated March 31, 1980 (P-80064), Fort St. Vrain In-service Inspection and Testing (Reactor Primary and Secondary Coolant Systems).
15. Los Alamos National Laboratory letter dated October 30, 1981 (Q-13:81:365).
16. Los Alamos National Laboratory letter dated November 2, 1981 (Q-13:81:369) (Proposed Agenda for a Meeting November 20, 1981).
17. Public Service Company letter dated November 9, 1981 (P-81285), Los Alamos National Laboratory Evaluation of Fort St. Vrain ISI Program.
18. Los Alamos National Laboratory letter dated December 10, 1981 (Q-13:81:420), Fort St. Vrain ISI Program Review Meeting.
19. Los Alamos National Laboratory letter dated January 5, 1982 (Q-13:82:5) (Review of the Public Service Company Proposed In-service Inspection Program).
20. Public Service Company letter dated March 29, 1982 (P-82061), Fort St. Vrain In-service Inspection and Testing (Response to the Recommendations of Los Alamos National Laboratory Report Q-13:82:5).
21. Los Alamos National Laboratory letter dated June 30, 1982 (Q-13:82:228) (Comments Regarding Public Service Company Response).
22. Meeting of July 29, 1982, between Nuclear Regulatory Commission, Los Alamos National Laboratory, Public Service Company, and their Consultants.
23. Public Service Company letter dated September 30, 1982 (P-82430), Fort St. Vrain In-service Inspection and Testing Program Additional Surveillance Requirements.