ATTACHMENT 2

PROPOSED MODIFICATIONS TO THE FORT ST. VRAIN

TECHNICAL SPECIFICATIONS

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.

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.

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.

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 eacn 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 non-operable 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.

5.2 PRIMARY COOLANT SYSTEM - SURVEILLANCE REQUIREMENTS

Applicability

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

Objective

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

Specification SR 5.2.1 - PCRV and PCRV Penetration Overpressure Protection Surveillance

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

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

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.

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.

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.

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.

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 cannot be tested. However, one rupture disc of each type assembly is visually examined to verify that the membrane is free of defects and that the knife blade remains sharp.

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

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

| Specification SR 5.2.2 - Tendon Corrosion and Anchor Assemblies

| Surveillance

The serviceability of the corrosion protection applied to and | the condition of the prestressing tendons shall be monitored in | accordance with paragraph 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.

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.

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.

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 calender year intervals during a vessel pressurization to operating pressure.

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

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. Re-inforcing steel is provided to control crack growth development with respect to size and spacing. Model testing has also shown that severely cracked vessels contain the normal working pressure for extended periods of time, as long as the effective prestressing forces are maintained.

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

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

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

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

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

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

Specification SR 5.2.5 - Liner Specimen Surveillance

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

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

Basis for Specification SR 5.2.5

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

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

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

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

Tests performed on this liner material (see FSAR Section 5.7.2.2) have indicated that no observable changes in material characteristics developed during an exposure to a fluence equivalent to the first five years of full power operation. Further, these tests demonstrated no significant damage after a fluence equivalent to 30 years of 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 5.2.16 - 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.

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.

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.

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

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.

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.

| <u>Specification SR 5.2.17 - Helium Circulator Pelton Wheels</u>

| DELETE SPECIFICATION SR 5.2.17 IN ITS ENTIRETY

| <u>Specification SR 5.2.18 - Helium Circulators Surveillance</u>

| 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.

1) 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.

<u>Basis for Specification SR 5.2.18</u>

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

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

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

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

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

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

Specification SR 5.2.22 - 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 5.2.25 - 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

| 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:

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.

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.

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

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.

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.

| 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.

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.

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

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

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

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

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

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

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.

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.

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.

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.

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.

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 relief.

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

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

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 valve 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.

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.

Specificaton 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.

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.

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.

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.

ATTACHMENT 3

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 determined 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. <u>EVALUATION AND CONCLUSIONS</u>

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.

ATTACHMENT 4

IMPLEMENTATION SCHEDULE

Included hereafter is a schedule which specifies for each proposed change to the Technical Specifications the latest date at which that change will have been implemented. The following notations are used:

T0 Indicates that the proposed Technical Specification change will be implemented before 90 days have elapsed following the formal approval date by the Nuclear Regulatory Commission.

T1 Indicates that the proposed Technical Specification change will 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 by the Nuclear Regulatory Commission

Otherwise, the proposed Technical Specification change will be implemented before the end of the first scheduled plant shutdown following 90 days from the formal approval date by the Nuclear Regulatory Commission.

T2 Indicates that the proposed Technical Specification change will be implemented before the beginning of fuel cycle 5.

TX Indicates that the proposed Technical Specification change will be implemented in the existing schedule of surveillance tests, following 90 days from the formal approval date by the Nuclear Regulatory Commission.

SR 5.1.2.a	T0	
SR 5.1.2.f	T1	
SR 5.1.2.g	T1	
SR 5.2.1.a	T2	
SR 5.2.1.b	TX	(Staggering of safety valve testing will be initiated following the next scheduled test.)
SR 5.2.1.c.1	TX	
SR 5.2.1.c.2	T2	(Same as SR 5.2.1.a.)
SR 5.2.1.c.3	TX	(At next scheduled SR 5.2.1.b.)
SR 5.2.2.a	TX	
SR 5.2.2.c	TX	(Same as SR 5.2.2.a)
SR 5.2.3.b	T0	
SR 5.2.4.b	T2	
SR 5.2.4.c	T2	
SR 5.2.5	As specified in proposed Technical Specification (fifth refueling cycle).	
SR 5.2.16.a	T0	
SR 5.2.16.c	T0	
SR 5.2.16.d	T1	
SR 5.2.16.e	T1	
SR 5.2.16.f	T1	
SR 5.2.18	TX	
SR 5.2.25	TX	(In conjunction with SR 5.2.22.)
SR 5.2.26	T1	
SR 5.2.27	T2	
SR 5.2.28.a	T2	
SR 5.2.28.b.1	T2	
SR 5.2.28.b.2	T1	
SR 5.2.28.b.3	T2	
SR 5.2.28.c	T2	
SR 5.3.1.b	T0	
SR 5.3.1.c	T1	
SR 5.3.2	T0	
SR 5.3.3	T0	
SR 5.3.4	T1	
SR 5.3.9	T1	(Initiation of safety valve staggered testing.)
SR 5.3.10	T1	
SR 5.3.11	T2	