FORT ST. VRAIN SAFETY PARAMETER DISPLAY SYSTEM

SAFETY ANALYSIS REPORT

CONTENTS:

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1.0 INTRODUCTION

A Safety Parameter Display System (SPDS) is being incorporated into the control room to provide the operators with overall plant safety status and warnings of degraded conditions significant to safety. The plant safety status is based on the status of five critical safety functions, each characterized by a set of parameters monitored by the SPDS.

The overall objective of plant safety is the protection of the health and safety of the general public by ensuring that potential radiation exposure is within established limits for all normal and postulated abnormal conditions. This objective is accomplished:

- by preserving the integrity of nuclear fuel, thereby preventing excessive fission product release to the reactor coolant system, and

- by preserving the integrity of the reactor coolant system, thereby preventing the release of radioactivity to the environment.

By displaying, monitoring, and alarming the status of critical safety functions in a single dedicated control board area, the operators are provided with knowledge concerning any challenge to the integrity of these fission product barriers. The SPDS thus provides an additional means of assuring that the overall plant safety objective is achieved.

2.0 PURPOSE

The purpose of this safety analysis is to justify that the parameters selected for the SPDS are sufficient to assess the status of the critical safety functions for a wide range of events, including symptoms of severe accidents. Five critical safety functions have been identified for the Fort St. Vrain reactor. They are:

- Reactivity Control (RC), which relates to heat generation;

- Primary Heat Removal (PHR), and

- Secondary Heat Removal (SHR), both of which relate to primary coolant circulation for heat removal from the core and heat transfer to the secondary coolant;

- Primary Coolant System Integrity (CSI), which relates to both core cooling and containment of fission products; and

- Radioactivity Control (RAC), which relates to fission product releases from the fuel or PCRV.

3.0 APPROACH

The principal function of the SPDS is to aid the control room personnel during normal, abnormal and emergency conditions to determine the safety status of the plant and to assess whether abnormal conditions warrant corrective action by the operators (SPDS Program Plan, Section 1.2).

In order to satisfy this function, the SPDS must be able to detect, in a timely manner, abnormal conditions that are significant to safety. Based on the overall plant safety objective, abnormal conditions significant to safety are considered to be those which, if left uncorrected, could allow or have already allowed fission products to escape from the fuel particles or from the PCRV.

To protect the integrity of these fission product barriers, two key operating safety limits have been established in Section 3 of the Technical Specifications: a reactor core safety limit, and a reactor vessel safety limit. They are: (1) limits on power-to-flow ratio to prevent elevated fuel particle temperature, thus preserving the integrity of the particle coatings, and (2) limits on reactor vessel pressure to prevent high concrete and metal stresses, thus preserving the integrity of the PCRV and the primary coolant pressure boundary.

Those parameters which can indicate that fuel particle temperature or PCRV pressure is unsafe or could become unsafe are also parameters which will indicate that abnormal conditions significant to safety are present. In addition, parameters which indicate that either type of fission product barrier has actually been breached (e.g., due to material defects unrelated to the safety limits) are also parameters which can indicate that abnormal conditions significant to safety are present. The combination of these two types of parameters will then provide a parameter set that is sufficient to satisfy the function of the SPDS. These parameters can then be logically grouped according to the critical safety functions for which they are most representative. This allows the safety status of the fission product barriers to be consolidated, thus enhancing timely and meaningful detection.

This safety analysis therefore focuses on determining those parameters which will in fact detect a real or potential challenge to the safety limits (Technical Specification SL 3.1 and SL 3.2) or an actual breach of the fission product barriers.

4.0 SUMMARY AND CONCLUSION

Implementation of the approach outlined in Section 3.0 is discussed in detail in Section 5.0. Based on the results of this analysis, the preliminary list of SPDS parameters included in the Fort St. Vrain SPDS Program Plan (Ref. PSC Letter P-84013 dated January 20, 1984) has been upgraded as indicated in Table 1.

A large number of selected SPDS parameters are associated with the critical safety functions of reactivity control, primary heat removal and secondary heat removal. These parameters provide for protection of fuel particle integrity by indicating abnormal conditions which could result in elevated fuel temperatures. The large thermal inertia of the Fort St. Vrain reactor, combined with its negative temperature coefficient of reactivity, results in core thermal transients which are relatively slow. Thus, the SPDS monitoring of parameters directly representative of power generation and normal core heat removal capability is sufficient for the timely identification of degraded plant conditions which could adversely affect fuel particle integrity. As a result, the corresponding SPDS parameters are function or symptom oriented rather than system or equipment oriented; they adequately characterize nuclear power generation, primary system heat removal capability, and secondary system heat removal, including normal circulator drive capability. The selected parameters also provide the information required for compliance with related Technical Specifications where prompt operator corrective action may be required to preserve the fuel particle coating safety limit. Any degraded condition significant to plant safety which could potentially result in elevated fuel temperatures, whatever the cause, will challenge at least one of the above functions and associated parameter(s). These degraded conditions will be detected by the SPDS and will be evaluated based on information provided by other existing plant instrumentation available in the control room.

The selected SPDS parameters associated with the critical safety function of primary coolant system integrity monitor those abnormal conditions, whatever the cause, which result in increased pressures which could become detrimental to the integrity of the PCRV. Some of these parameters, together with the remaining SPDS parameters associated with the critical safety function of radioactivity control, monitor the integrity of the two types of barriers to fission product release (fuel particle coatings, and the reactor coolant system boundary, which includes the PCRV, PCRV liner, PCRV penetrations, and steam generator economizer/evaporator/superheater and reheater sections).

Because of the function or symptom oriented nature of the selected parameters, the SPDS is capable of detecting a wide range of events that include symptoms of severe accidents, since these events would necessarily result in the degraded performance of at least one of the critical safety functions. The SPDS capability has been verified by reviewing those events (accidents and incidents identified in Section 14 of the FSAR) which have the potential for releasing abnormal quantities of radioactivity. It was determined that each event would significantly affect at least one of the selected SPDS parameters. In many cases, these selected SPDS parameters are also primary indicators of the event.

5.0 ANALYSIS

5.1 Discussion

5.1.1 Plant Technical Specifications

The plant Technical Specifications include two safety limits: a reactor core safety limit and a reactor vessel safety limit. Operation within these limits ensures that conditions do not result in an uncontrolled or unplanned release of radioactivity. The parameters associated with these safety limits are of significance to safety and therefore have been evaluated for inclusion in the SPDS.

The plant Technical Specifications also include limiting safety system settings (LSSS) and performance oriented limiting conditions for operation (LCO) in support of the safety limits or in support of safety analyses for worst case accidents. The parameters associated with these LSSS's or LCO's are also of significance to safety and have been considered for the SPDS.

The plant Technical Specifications described above include provisions for required operator corrective action to protect the barriers to fission product release. In several instances, the time allowed for the specified corrective action is relatively short, requiring that the operator quickly be made aware that such action is needed. Therefore, the timeliness of SPDS response to the corresponding abnormal conditions is also a consideration in the choice of parameters to be displayed.

5.1.2 RESPONSE TIME

The SPDS response time for an individual parameter is the sum of the SPDS updating time, the time required for calculation, and the alarm processing time. The SPDS response times for the various selected parameters are summarized in Table 3.

The SPDS response time for individual parameters has been reviewed in light of the transients for various accidents evaluated in the FSAR, as well as allowable times for successful operator corrective action specified in the plant Technical Specifications.

The update frequency is the same as that used for the plant computer. Therefore, this 5 second update has been demonstrated to be acceptable for plant operation.

5.1.3 Fuel Temperature Control

The nuclear fuel used at the Fort St. Vrain Nuclear Generating Station is in the form of pellets composed of coated particles bonded in a graphite matrix. The fuel pellets are stacked in fuel channels in graphite fuel blocks. These fuel blocks also include cooling channels for removal of fission reaction generated heat. The fuel blocks are arranged to form the core of the reactor. Cooling is achieved by forced circulation of pressurized helium. The whole reactor coolant system is contained in a prestressed concrete reactor vessel (PCRV) which is housed in a vented reactor building.

During normal plant operation, radioactivity released from the fuel circulates in the reactor coolant system. Part of the radioactive material plates out on cooler metal surfaces of the reactor coolant system components. Accidents were postulated for Fort St. Vrain which result in release of the reactor coolant to the environment. They are (1) the Maximum Credible Accident (FSAR Section 14.8) wherein the entire primary coolant circulating inventory is carried out of the PCRV and is released to the atmosphere through the reactor plant ventilation system, and (2) Design Basis Accident No. 2 (FSAR Section 14.11) wherein a PCRV rapid depressurization causes the entire primary coolant circulating inventory and part of the plateout to be carried out of the PCRV and the reactor building. For the radiological consequences of these two depressurization accidents to remain within acceptable limits, it is necessary to establish maximum limits for the radioactivity present in the reactor coolant system. Technical Specification LCO 4.2.8 specifies these limits for primary coolant gaseous and plateout activity levels.

Since the reactor coolant activity limits specified in LCO 4.2.8 are only a fraction of the activity generated by the fission reaction and retained within the fuel particle coating, it is of significant importance to plant safety that the integrity of the fuel particle coatings be preserved.

This is achieved by ensuring that the fuel temperature remains within acceptable limits by controlling the balance between heat generation and heat removal. Heat generation is controlled by controlling core reactivity to adjust the rate of fission reactions.

Two major plant systems are involved in the removal of core generated heat during plant operation at power, shutdown, and also following all postulated accidents except loss of forced circulation cooling accidents. They are (1) the primary reactor coolant system which includes four helium circulators driven by steam turbines for operation at power and driven by steam or water turbines for other modes of operation, and (2) the secondary coolant system which includes two steam generators, each one composed of six modules with one economizer/evaporator/superheater section and a reheater section. Cold reheat steam provides the motive power to the helium circulator steam turbine drives during power operation.

The primary coolant system transfers most of the core generated heat to the steam generators. Primary coolant at core outlet temperature flows over the reheater section of the steam generators, then over the EES section. Because of this arrangement, hot reheat steam temperature is sensitive to variations in primary coolant core outlet temperature.

A small fraction of the total heat generated is also transferred to the PCRV cooling water system, or dissipated in losses to the reactor building atmosphere. For those plant conditions where forced circulation cooling is available, the significance of the PCRV cooling water system to fuel temperature control is negligible.

The Fort St. Vrain reactor is quite tolerant to degraded core cooling conditions, up to and including the postulated permanent loss of all forced circulation cooling analyzed as Design Basis Accident No. 1 (FSAR Section 14.10). The basis to Technical Specification LCO 4.2.18 specifies the maximum allowable time between loss of forced circulation and initiation of PCRV depressurization, which is the first required operator action if forced circulation has not been restored in the meantime. In no case is that time period less than two hours. Reactor core decay heat removal is accomplished by the PCRV liner cooling water system. A DBA-1 occurring simultaneously with a loss of PCRV liner cooling water has been analyzed (FSAR Section D.2.3), and it was determined that up to 30 hours would be available, from accident initiation, to restore an adequate supply of cooling water. Therefore, PCRV cooling capability is not a critical safety function in assessing the plant safety status.

Based on these characteristics of the Fort St. Vrain reactor, the reactor coolant system parameters will provide adequate information to the operators to warn of degraded plant conditions. Nonetheless, additional parameters are considered appropriate for the SPDS which relate to the secondary coolant system. These parameters will also provide early indication of degraded conditions which may result in elevated fuel temperatures.

There are numerous parameters indicative of abnormal conditions affecting primary and secondary coolant systems with potentially significant impact on fuel temperature control. These parameters are grouped to characterize two critical safety functions, namely primary heat removal and secondary heat removal.

5.1.4 Primary Coolant System Integrity

The importance of primary coolant system integrity to plant safety is two-fold. By preventing loss of primary coolant inventory, it participates in the fuel temperature control function and it prevents excessive release of radioactivity.

The primary coolant system of the Fort St. Vrain reactor is housed in a prestressed concrete reactor vessel which has the dual function of primary and secondary reactor coolant pressure boundary. The PCRV includes various equipment and access penetrations. These penetrations are equipped with two closures: a primary closure and a secondary closure. The reactor coolant system is contained by the primary closures. During normal operation, the interspace between the penetration primary and secondary boundaries is pressurized with purified helium at a slightly higher pressure than the primary coolant. The structural integrity of the PCRV depends on the control of the pressure therein.

The structural integrity of the PCRV also depends on continued prestressing and liner cooling. The effects of degraded conditions concerning these two functions would be slow and, therefore, need not be considered for the SPDS. Other plant instrumentation will provide the operators with timely warning to take any required corrective action.

5.1.5 Radioactivity Control

The release of radioactive fission products is controlled by two types of barriers: the fuel particle coatings and the reactor coolant system boundary. The integrity of these barriers is indicated by monitoring the primary coolant activity, and the reactor coolant system boundary which includes the PCRV, PCRV liner, certain PCRV penetrations, and the steam generators.

5.1.6 Operating Modes

The selected SPDS parameters are adequate to monitor the plant safety status at the various levels of power operation (startup, low power, and power), as determined by the position of the interlock sequence switch, and during the various modes of plant operation (fuel loading, off, and run), as determined by the position of the reactor mode switch. Since the normal ranges of parameter evolution under these various conditions vary, the setpoints for SPDS warning and alarms will be automatically adjusted as appropriate for each individual parameter. Under certain operating conditions, some of the selected parameters are not relevant for the determination of the plant safety status and these parameters will not be alarmed by the SPDS. Table 2 delineates applicable parameters which are meaningful and relevant for specific operating modes and power levels.

5.2 Results

On the basis of the above discussion, parameters have been selected for the Fort St. Vrain SPDS which characterize each of the five identified critical safety functions.

In the following sections, these parameters are identified and the justification for their selection is discussed.

5.2.1 Reactivity Control

5.2.1.1 Selected SPDS Parameters

The following parameters have been selected for the Fort St. Vrain SPDS as they most closely relate to the reactivity control function.

- Average Neutron Power ------------- (Table 1, Item #1)
- Neutron Flux Rate of Change ------- (Table 1, Item #2)
- Primary Heat Balance Power -------- (Table 1, Item #3)
- Secondary Heat Balance Power ------ (Table 1, Item #4)

The justification for the selection of each above parameter is discussed in Section 5.2.1.2 below.

5.2.1.2 Justification for Parameter Selection

5.2.1.2.1 Average Neutron Power

The thermal power generated by fission reaction is a function of the neutron flux which itself depends on core reactivity. The core power is controlled by adjusting the position of the control rods and, therefore, by controlling core reactivity. Any abnormal condition resulting in a net change in core reactivity will be reflected by a change in neutron power. Indication of neutron power is provided at all power levels and reactor modes except "refueling" and "off" by utilizing the linear power channels.

To support the core safety limit, a limiting safety system setting of 140 percent of rated thermal power is specified for reactor scram in LSSS 3.3. The Technical Specification basis indicates that this single value is sufficient for the plant because the negative temperature coefficient of reactivity and large thermal capacity of the reactor core restrict the transient increases in fuel and helium temperature to acceptable limits. In practice, control rod configuration alters the radial core power distribution so that the neutron flux levels seen by the power range detectors may not directly represent the true core power level (FSAR Section 7.3.1.2.1). This detector decalibration phenomenon requires adjustments to each channel trip setpoint as necessary to assure that the trip occurs before the actual reactor power exceeds 140 percent of rated power. There is no other parameter related to reactivity control which would automatically scram the reactor during plant operation at power. Since the power range channels measure power levels as low as 1.5 percent of rated power, however, they also detect abnormal conditions significant to safety with respect to reactivity control while the reactor is operating at low power.

An automatic scram is utilized to protect fuel particle integrity in the event of a large increase in power. Therefore, the 20 second response time is adequate to notify the operator that additional action may be required.

5.2.1.2.2 Neutron Flux Rate-of-Change

Neutron flux rate-of-change has been included in the SPDS to provide indication of reactivity control abnormalities for all reactor modes. Startup channels I and II and Wide Range Channel III have been selected as inputs to the SPDS. Startup channels I and II have a range of 1 count per second to 2E+05 counts per second (approximately 1E+02 % power). Wide Range channel III has a range of 1E-06 to 1E+03% power. Therefore the SPDS will have the capability to monitor neutron flux from refueling to power range conditions. All of the neutron flux rate of change indicators have a range from -1 to 0 to +7 decades per minute (dpm), therefore abnormal changes in reactivity are monitored effectively for all reactor modes.

As for average neutron power, neutron flux rate-of-change also initiates an automatic scram. Therefore, a 20 second response time is adequate to alert the operator to the fact that protective action should have occurred, and operator action may be required.

5.2.1.2.3 Primary Heat Balance Power and Secondary Heat Balance Power

The fuel temperature is determined on the basis of a thermodynamic balance between the fission generated power in the core, the power removed by the primary system, and the power transferred to the secondary system. Significant imbalances between heat generation and heat removal capability could result in elevated fuel temperatures with potential damage resulting to the fuel particle coating. Therefore, an indication of reactivity control and the overall plant safety status can be provided by a direct comparison of the average neutron power, primary heat balance power, and secondary heat balance power. Under degraded heat removal conditions, the reactor power would have to be reduced at least to a level where heat generation would not exceed the most restrictive heat removal capability of the primary system or secondary system.

Therefore, Primary Heat Balance Power and Secondary Heat Balance Power are two parameters related to the reactivity control function which have been selected for the SPDS.

The 1 minute, 5 second and 5 minute, 5 second SPDS response times for the primary and secondary heat balance powers, respectively, are adequate since these parameters are best suited for the monitoring of slowly evolving power imbalances which could occur during normal plant operation. Therefore, it is not appropriate to alarm these parameters.

5.2.2 Primary Heat Removal

5.2.2.1 Selected SPDS Parameters

The following parameters have been selected for the SPDS as they closely relate to the primary heat removal function:

- Power-to-Flow Ratio -------------------------(Table 1, Item #5)
- Primary Helium Flow ------------------------(Table 1, Item #6)
- Core Average Outlet Temperature--------------(Table 1, Item #7)
- Maximum Region Outlet Temperature Mismatch --(Table 1, Item #8)
- Average Circulator Inlet Temperature --------(Table 1, Item #9)

The justification for the selection of each parameter is discussed in Section 5.2.2.2 below.

5.2.2.2 Justification for Parameter Selection

5.2.2.2.1 Power-to-Flow Ratio

A core safety limit is specified in Technical Specification SL 3.1 to prevent conditions which may result in fuel temperature levels where migration of the fuel kernel through the particle coating could occur. This is achieved by establishing limits of allowable operating time as a function of the reactor power-to-flow ratio. This ratio is the ratio of percent reactor thermal power to percent primary coolant flow. The area under the curve of Figure 3.1-2 of the Technical Specifications establishes the maximum power-to-flow ratio which can be sustained indefinitely as a function of reactor power level. Power-to-flow ratio values in excess of the limits of Figure 3.1-2 are acceptable for an integrated time of 100 hours over the life of the fuel; for such cases, SL 3.1 allows 30 minutes for the operator to return the power-to-flow ratio within acceptable limits. The allowable integrated time for operation with power-to-flow ratio values exceeding 1.17 decreases from 100 hours to about 2 minutes for a power-to-flow ratio of 2.5, according to a curve included in Figure 3.1-1 of the Technical Specifications. For a power-to-flow ratio exceeding 1.17 but less than or equal to a value of 2.5, SL 3.1 allows 2 minutes for corrective action to bring the power-to-flow ratio back to a value less than 1.17.

High values of the power-to-flow ratio would result from increases in power generation without compensating increase in reactor coolant flow, or from decreases in reactor coolant flow without compensating decreases in power generation. There is a broad range of abnormal power and/or flow conditions where exceeding the power-to-flow limits specified in SL 3.1 will not result in exceeding the corresponding limiting safety system settings specified in Technical Specification LSSS 2.3. For those cases, no automatic protective action will take place, and therefore, it is necessary that the operator be warned that corrective action is required in accordance with SL 3.1. Therefore, the power-to-flow ratio is a parameter which has been selected for the SPDS to provide additional warning that abnormal conditions are present which could become significant to safety.

The SPDS response time for the parameters which characterize the primary system heat removal function are evaluated against the shortest applicable time allowed in the plant Technical Specifications for successful operator corrective action in case defined limits are exceeded. These specified times reflect the severity of the transient parameter behavior (the more severe the condition, the shorter the allowable time for corrective action), and their values were conservatively determined so that the core safety limit would not be exceeded. Under postulated accident conditions, the transient behavior of the parameters under consideration is sustained over time, and no short lived transients of significance to safety which require operator action have been identified which might not be detected by the SPDS. Therefore, the SPDS response time for any parameter under consideration will not unduly restrict the timely detection of an abnorman condition nor the operator's ability to take the applicable corrective action within the specified time.

5.2.2.2.2 Primary Helium Flow

This parameter is indicative of any abnormal condition which may result in a general decrease of reactor coolant flow and which, if not corrected or compensated by a decrease in power, could result in elevated fuel temperatures. Therefore, reactor coolant flow is a parameter which has been selected for the SPDS to provide early indications of abnormal conditions significant to safety.

A 1 minute, 25 second SPDS response time for primary helium flow, which is a calculated parameter, is adequate since the effect of reduced flow on the power-to-flow ratio will be monitored as indicated above. Further, Technical Specification LCO 4.2.9 allows for 15 minutes to return to operation within specified limits before initiating shutdown, which is large compared to a 1 minute, 25 second SPDS response time.

5.2.2.2.3 Reactor Coolant Temperatures (Core Average Outlet Temperature, Maximum Region Outlet Temperature Mismatch, and Average Circulator Inlet Temperature).

The reactor coolant flow is distributed between the various core refueling regions by manually adjusting the position of core inlet orifice valves to yield approximately uniform core outlet temperatures during power operation.

Technical Specification LCO 4.1.7 includes limiting values of core region outlet temperature mismatch for individual refueling regions. The temperature mismatch is the difference between the region outlet temperature and the core average outlet temperature, the latter being defined in Technical Specification Definition 2.20 as the arithmetic average of the individual region outlet temperatures.

For a core average outlet temperature less than 950 degrees F, a maximum temperature mismatch of 400 degrees F is allowed for all regions, and the conditions of LCO 4.1.9 must be met.

For a core average outlet temperature of 950 degrees F or more, two maximum temperature mixmatches are specified in LCO 4.1.7: (1) a maximum temperature mismatch of 50 degrees F (Mismatch B) applicable to the 9 regions whose orifices are most fully closed and to all regions with control rods inserted more than 2 feet; and (2) a variable maximum temperature mismatch (Mismatch A), applicable to the remaining regions, which is a function of the average circulator inlet temperature and the average temperature rise from circulator inlet to core outlet.

LCO 4.1.7 allows 24 hours for successful corrective action if the specified mismatch limits are exceeded by less than 50 degrees F, and 2 hours for successful corrective action if the limits are exceeded by 50 degrees F but less than 100 degrees F. Should the limits be exceeded by 100 degrees F or more , immediate orderly shutdown is required. The basis for LCO 4.1.7 indicates that the specified mismatch limits are more conservative than those used to develop SL 3.1, and that the times allowed for successful corrective action at temperatures exceeding the limits represent conditions remaining well below the core safety limits.

Because LCO 4.1.7 supports the validity of assumptions upon which SL 3.1 is based, and because relatively fast corrective actions may be required, the various parameters involved are significant to plant safety and have been selected for the SPDS. They are: (1) the core average outlet temperature required for the determination of the applicable LCO requirement, and for the calculation of the average temperature rise utilized to determine the allowable Mismatch A; (2) the average circulator inlet temperature utilized to determine the allowable value of Mismatch A; and (3) the maximum region outlet temperature mismatch corresponding to the region requiring the most immediate operator attention. Since two different limits may apply depending on the regions operating configuration (Mismatch A or Mismatch B), two maximum region outlet temperature mismatches have been selected for the SPDS: (a) the highest mismatch B for the 9 regions whose orifices are most fully closed and for all the regions with control rods inserted more than 2 feet; and (b) the highest mismatch A for the remaining regions.

In addition to the core outlet temperature mismatch limits specified in LCO 4.1.7, Technical Specification LCO 4.1.9 specifies limits for the applicable core region temperature rise or minimum percent circulator mass flow. The basis for LCO 4.1.9 indicates that a maximum core region helium coolant temperature rise or minimum percent circulator mass flow as a function of thermal power at lower power levels, and therefore low reactor coolant flowrate conditions, is specified to prevent laminar flow stagnation in any coolant channel, and subsequent degradation of heat transfer characteristics which could result in excessive fuel temperatures. LCO 4.1.9 specifies various limits for region temperature rise or for reactor coolant flow, as applicable, which depend on the primary system helium density and on the orifice valve configuration (set for equal coolant channel flow in all the regions or orifice valves set at any other position). LCO 4.1.9 allows 15 minutes for successful corrective action, should a specified region temperature rise limit be exceeded or should minimum total circulator mass flow rate not be available.

The maximum region temperature rise, corresponding to the region which should deserve the most immediate operator attention, is evaluated using the average circulator inlet temperature, and the individual refueling region outlet temperatures. The total circulator mass flow rate, the primary heat balance power, and the primary system pressure are also selected parameters, so that all information required for compliance with the requirements of LCO 4.1.9 can be derived from SPDS parameters.

A 1 minute, 25 second SPDS response time for the core average outlet temperature, the maximum region outlet temperature mismatch and for the average circulator inlet temperature is adequate in light of the allowable times specified in LCO 4.1.7 and LCO 4.1.9.

5.2.3 Secondary Heat Removal

5.2.3.1 Selected SPDS parameters

The following parameters have been selected for the SPDS as they closely relate to the secondary heat removal function:

- Feedwater Flow -----------------------------(Table 1, Item #10)
- Main Steam Temperature ---------------------(Table 1, Item #11)
- Main Steam Pressure ------------------------(Table 1, Item #12)
- Hot Reheat Steam Temperature ---------------(Table 1, Item #13)
- Hot Reheat Steam Pressure ------------------(Table 1, Item #14)
- Steam Jet Air Ejector Activity -------------(Table 1, Item #15)

Where applicable, the various secondary system parameters identified above are to be monitored by the SPDS for each loop. This is more conservative than monitoring on a total plant basis, since the SPDS will be more sensitive to degraded conditions affecting the secondary system and, therefore, will be capable of providing earlier warnings to the operators.

The justification for the selection of each parameter is discussed in Section 5.2.3.2 below.

5.2.3.2 Justification for Parameter Selection

5.2.3.2.1 Feedwater Flow, Main Steam Temperature, and Main Steam Pressure

Feedwater flow, main steam temperature, and main steam pressure have been selected for the SPDS because they are sensitive to perturbations in the system. Deviation from their control range will provide early indication of degraded system conditions which could eventually result in elevated fuel temperatures. In addition, main steam pressure is indicative of loss of system integrity which could result in loss of circulator steam turbine motive power.

5.2.3.2.2 Hot Reheat Steam Temperature

A limiting safety system setting is specified in Technical Specification LSSS 3.3 for the hot reheat steam temperature. As indicated previously, because of the steam generator configuration and primary coolant flow path, reheat steam temperature is quite sensitive to variations in the primary coolant reactor outlet temperature (i.e., reactor thermal power). This parameter is used for plant protective system scram input as well as for normal control of reactor power. Increases in hot reheat steam temperature would indicate a variety of abnormal conditions including those resulting in increased reactor outlet temperature and those resulting in or from decreased reheat steam flow. Therefore, this parameter has been selected for the SPDS.

5.2.3.2.3 Hot Reheat Steam Pressure

Hot reheat steam pressure has also been selected for the SPDS. This parameter provides a scram input to the plant protective system. SPDS monitoring of hot reheat steam pressure would warn the operator of abnormal decreases, indicative of loss of system integrity which, as with main steam pressure, could also result in loss of circulator steam turbine motive power.

5.2.3.2.4 Steam Jet Air Ejector Activity

The presence of radioactivity at the steam jet air ejector discharge may be indicative of a leak in a steam generator reheat section, since the reheaters operate at a pressure lower than the primary coolant. In addition to indicating a degraded primary coolant pressure boundary at the steam generator, steam jet air ejector activity also provides an early warning that actions affecting the secondary coolant system may be required to isolate the leak, with subsequent impact on primary coolant circulation and core heat removal capability. Therefore, this parameter has been selected for the SPDS to characterize the secondary heat removal function. Since the effluent is released to the plant vent, the stack radiation monitors could also provide an indication of degraded radioactivity control function.

5.2.4 Primary Coolant System Integrity

5.2.4.1 Selected SPDS Parameters

The following parameters have been selected for the Fort St. Vrain SPDS as they closely relate to the critical safety function of preserving the primary coolant system integrity:

- Primary Coolant Pressure ---------------(Table 1, Item #16)
- Primary Coolant Moisture, --------------(Table 1, Item #17)
- Circulator and Steam Generator
  Penetration Interspace Pressure --------(Table 1, Item #18)

The justification for the selection of each parameter is discussed in Section 5.2.4.2 below.

5.2.4.2 Justification for Parameter Selection

5.2.4.2.1 Primary Coolant Pressure

Technical Specification SL 3.2 specifies a safety limit applicable to the primary coolant pressure. This safety limit is the PCRV Reference Pressure of 845 psig which should not be exceeded in order to provide conservative design margins to protect the integrity of the PCRV. Abnormal increases in primary coolant pressure represent conditions significant to plant safety which should be detected by the SPDS.

Also, accidents have been postulated which result in a depressurization of the reactor coolant system. The primary coolant heat transfer characteristics are a function of primary coolant pressure and deteriorate as the pressure decreases. Technical Specification LSSS 3.3 specifies a limiting safety system setting for reactor scram upon low primary system pressure. This setting is programmed with reactor power (by modifying its value as a function of the average circulator inlet temperature) so that fission heat production will be automatically interrupted as early as possible upon detection of an abnormally low reactor coolant pressure, thereby limiting the fuel temperature increase resulting from degraded heat transfer conditions.

Additionally, an abnormally low primary coolant pressure may be an indication of loss of primary coolant system integrity which, by itself, is an event of significant importance to safety.

Therefore, the primary coolant pressure is a parameter which has been selected for the SPDS since it indicates both conditions which could challenge the integrity of the reactor coolant system boundary and conditions wherein that integrity has already been degraded.

The 20 second SPDS response time for primary system pressure is also adequate for depressurization accidents. In the case of the Maximum Credible Accident (FSAR Section 14.8), it takes about 20 minutes for the pressure to decrease to half its nominal value, and a PPS low pressure scram would take place at about 90 seconds. In the case of a rapid depressurization/blowdown accident (DBA2-FSAR Section 14.11), PCRV depressurization is complete in about 2 minutes and the low pressure scram would occur at about 2 seconds. It should further be noted that DBA2 is considered to be an incredible event, and that no operator action is anticipated.

5.2.4.2.2 Primary Coolant Moisture

The most likely cause for a significant pressure increase in the primary system would be water or steam ingress. In addition to the effect of moisture on primary system pressure, moisture at elevated temperature would also react with the core and core support graphite components and result in carbon transport (LCO 4.2.10). Also, at lower temperatures (below 1200 degrees F), there is a need to prevent corrosion of metals in the primary coolant system and to limit oxidation of burnable poison material in the core to acceptable levels. Even though these phenomena would only affect the reactor in the long range, it is desirable to minimize moisture ingress in the the primary system.

Technical Specification LSSS 3.3 includes a limiting safety system setting for reactor scram, loop shutdown and steam/water dump upon high moisture in the primary coolant system. Therefore, primary coolant moisture is a parameter significant to plant safety which has been selected for the SPDS. It would warn the operator of moisture ingress whatever its source. This parameter also monitors the integrity of the steam generator EES which is a primary coolant system boundary.

With the respect to moisture ingress accidents in the primary system, the worst postulated leak, due to the rupture of one steam generator EES subheader (FSAR Sections 14.5.2 and 14.5.3) would result in PPS protective action in about 9 seconds upon high moisture signal. If this action did not take place, PPS protective action would be initiated at about 95 seconds on a high primary coolant pressure signal. The moisture is sampled throughout one minute, then displayed to give a meaningful value. Rate of change for moisture is included to track the trend of primary coolant moisture and may be evaluated every 15 seconds. Due to the 1 minute, 25 second SPDS response time for primary coolant moisture, the operator will be alerted of a worst postulated steam inleakage accident by moisture rate of change or by an abnormally high primary coolant pressure which has a 20 second response time. For slow moisture inleakage which would not significantly affect primary system pressure, the 1 minute, 25 second SPDS response time for primary coolant moisture is adequate to provide timely warning to the operator.

5.2.4.2.3 Penetration Interspace Pressure

Technical Specification SL 3.2 includes a safety limit to provide conservative design margins to protect the integrity of the prestressed concrete reactor vessel. This safety limit applies to the PCRV penetration interspace pressure which is not to exceed the Reference Pressure value of 845 psig. Therefore, penetration interspace pressure is a parameter significant to plant safety which has been selected for the SPDS.

Only those penetrations with the potential for over-pressurization are considered (i.e. steam generator and circulator penetrations). Also, exceeding the corresponding plant protective system setpoint (810 psig or lower, per LCO 4.4.1) in any one of these penetration interspaces needs to be alarmed. This is sufficient, since penetration overpressure protection is provided in addition to automatic PPS action to limit the pressure buildup.

The 5 second SPDS update time is adequate for the circulator penetration interspace pressure. In the case of a service line rupture in such a penetration (FSAR Section 5.8.2.5.5), the PPS takes protective action at a pressure of about 757 psig. Overpressure protection takes place at a pressure of 840 psig, after approximately 60 gallons of water have leaked in. Considering the nominal bearing water flowrate of 100 gpm leaking in the penetration, it would take about 16 seconds to reach the PPS setpoint and actuate the automatic overpressure protection devices, which is required as part of the event mitigation. The 5 second SPDS response time after the PPS setpoint is reached is therefore also adequate for the operator to verify that the required automatic protection actions have taken place.

In the case of a worst postulated pipe rupture in a steam generator penetration, it is estimated that the time interval between the PPS protective action and actuation of the overpressure protection would be of the same order of magnitude as the 5 second SPDS response time. Therefore, this response time is also adequate for the operator to verify that the required automatic protective actions have taken place.

5.2.5 Radioactivity Control

5.2.5.1 Selected SPDS Parameters

The following parameters have been selected for the Fort St. Vrain SPDS as they most closely relate to the radioactivity control function:

- Primary Coolant Activity ----------------(Table 1, Item #19)
- Reactor Plant Exhaust Stack Activity ----(Table 1, Item #20)

The justification for the selection of each above parameter is discussed in Section 5.2.5.2 below.

5.2.5.2 Justification for Parameter Selection

5.2.5.2.1 Primary Coolant Activity

The most significant amount of radioactivity from the fission reaction is normally retained within the fuel particle coating. Part of it, however, escapes the fuel particle coating into the primary coolant system. As outlined previously, LCO 4.2.8 specifies maximum acceptable limits for primary coolant circulating activity, so that the radiological consequences of postulated depressurization accidents remain within the results of the FSAR analysis and within the limits of applicable regulations.

Therefore, primary coolant activity is a parameter of importance to plant safety, as it may indicate failure of fuel particles beyond the normally acceptable rate, and it has been selected for the SPDS.

Increases in primary coolant activity are expected to be slow and will be sustained over time, so that a 1 minute 25 second SPDS response time is adequate to identify degrading conditions.

5.2.5.2.2 Reactor Plant Exhaust Stack Activity

There are potential leak paths for radioactivity out of the reactor coolant system which include auxiliary systems, penetration closures, or instrument lines which penetrate the PCRV. Such radioactive leakage would eventually be collected by the ventilation system and would be filtered, prior to release, through the reactor plant exhaust stack. However, in order for the operator to take corrective action in case of abnormal or unplanned radioactivity release from the plant, radioactivity in the plant stack is a parameter which has been selected for the SPDS.

TABLE 1

FORT ST. VRAIN SPDS RECOMMENDED PARAMETER LIST

	Parameter	Critical Safety Function(*)	Comments
1.	Average Neutron Power	RC	
2.	Neutron Flux Rate of Change	RC	
3.	Primary Heat Balance Power	RC	
4.	Secondary Heat Balance Power	RC	
5.	Power-To-Flow Ratio	PHR	
6.	Primary Helium Flow	PHR	Total
7.	Core Average Outlet Temperature	PHR	Tech. Spec. Def. 2.20
8.	Max Region Outlet Temperature Mismatch	PHR	A & B (LCO 4.1.7)
9.	Average Circulator Inlet Temperature	PHR	
10.	Feedwater Flow	SHR	Each Loop
11.	Main Steam Temperature	SHR	Each Loop
12.	Main Steam Pressure	SHR	Each Loop
13.	Hot Reheat Steam Temperature	SHR	Each Loop
14.	Hot Reheat Steam Pressure	SHR	Each Loop
15.	Steam Jet Air Ejector Activity	SHR	
16.	Primary Coolant Pressure	CSI	
17.	Primary Coolant Moisture	CSI	
18.	Circulator and Steam Generator Penetration Interspace Pressure	CSI	Alarm High Only
19.	Primary Coolant Activity	RAC	
20.	Reactor Plant Exhaust Stack Activity	RAC	

(*) Critical Safety Functions

CSI = Reactor Coolant System Integrity
PHR = Primary Heat Removal
RAC = Radioactivity Control
RC = Reactivity Control
SHR = Secondary Heat Removal

TABLE 2

Fort St. Vrain SPDS Applicable Parameters
for Specific Operating Modes and Power Levels

REACTOR MODE SWITCH					
PARAMETER DESCRIPTION	FUEL LOADING	OFF	RUN — INTERLOCK SEQUENCE SWITCH		
			STARTUP	LOW POWER	POWER
1. Average Neutron Power			X	X	X
2. Neutron Flux Rate-of-Change	X	X	X	X	X
3. Primary Heat Balance Power			X	X	X
4. Secondary Heat Balance Power					X
5. Power-to-Flow Ratio				X	X
6. Primary Helium Flow	X	X	X	X	X
7. Core Average Outlet Temp	X	X	X	X	X
8. Region Outlet Temp Mismatch			X	X	X
9. Average Circ Inlet Temp	X	X	X	X	X
10. Feedwater Flow				X	X
11. Main Steam Temp					X
12. Main Steam Pressure					X
13. Hot Reheat Steam Temp			X	X	X
14. Hot Reheat Steam Pressure			X	X	X
15. Steam Jet Air Ejector Activity	X	X	X	X	X
16. Primary Coolant Pressure	X	X	X	X	X
17. Primary Coolant Moisture		X	X	X	X
18. Circ and Stm Gen Penet Pressure	X	X	X	X	X
19. Primary Coolant Activity	X	X	X	X	X
20. Stack Activity	X	X	X	X	X

TABLE 3

FORT ST. VRAIN SPDS RESPONSE TIME

	PARAMETER	TYPE	PARAMETER UPDATE FREQUENCY	ALARM PROCESSING TIME	SPDS RESPONSE TIME
1.	Average Neutron Power	Calculated	5 sec.	15 sec.	20 sec.
2.	Neutron Flux Rate-of-Change	Analog	5 sec.	15 sec.	20 sec.
3.	Primary Heat Balance Power	Calculated	1 min.	no alarm	1 min. 5 sec.
4.	Secondary Heat Balance Power	Calculated	5 min.	no alarm	5 min. 5 sec.
5.	Power-to-Flow Ratio	Analog Digital	5 sec. 5 sec.	15 sec. 1 sec.	20 sec. 5 sec.
6.	Primary Helium Flow	Calculated	1 min.	15 sec.	1 min. 25 sec.
7.	Core Average Outlet Temperature	Calculated	1 min.	no alarm	1 min. 25 sec.
8.	Maximum Region Outlet Temperature Mismatch	Calculated	1 min.	15 sec.	1 min. 25 sec.
9.	Average Circulator Inlet Temperature	Calculated	1 min.	15 sec.	1 min. 25 sec.
10.	Feedwater Flow	Analog	5 sec.	15 sec.	20 sec.
11.	Main Steam Temperature	Analog	5 sec.	15 sec.	20 sec.
12.	Main Steam Pressure	Analog	5 sec.	15 sec.	20 sec.
13.	Hot Reheat Steam Temperature	Analog	5 sec.	15 sec.	20 sec.
14.	Hot Reheat Steam Pressure	Analog	5 sec.	15 sec.	20 sec.
15.	Steam Jet Air Ejector Activity	Calculated	1 min.	15 sec.	1 min. 25 sec.
16.	Primary Coolant Pressure	Analog	5 sec.	15 sec.	20 sec.
17.	Primary Coolant Moisture	Calculated	1 min.	15 sec.	1 min. 25 sec.
18.	Circ. & Steam Generator Penetration Pressure	Digital	5 sec.	1 sec.	5 sec.
19.	Primary Coolant Activity	Calculated	1 min.	15 sec.	1 min. 25 sec.
20.	Stack Activity	Calculated	1 min.	15 sec.	1 min. 25 sec.