VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 13, 2011

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 11-594
SPS-LIC/CGL R0
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
ANNUAL SUBMITTAL OF TECHNICAL SPECIFICATIONS BASES CHANGES
PURSUANT TO TECHNICAL SPECIFICATION 6.4.J

Pursuant to Technical Specification 6.4.J, "Technical Specifications (TS) Bases Control Program," Dominion hereby submits changes to the Bases of the Surry TS implemented between October 1, 2010 and September 30, 2011.

Bases changes to the TS (that were not previously submitted to the NRC as part of a License Amendment Request) were reviewed and approved by the Facility Safety Review Committee. It was determined that the changes did not require a change to the TS or license, or involve a change to the UFSAR or Bases that required NRC prior approval pursuant to 10CFR50.59. These changes have been incorporated into the TS Bases. A summary of these changes is provided in Attachment 1.

TS Bases changes that were submitted to the NRC for information along with the associated License Amendment Request transmittals, submitted pursuant to 10CFR50.90, were also reviewed and approved by the Facility Safety Review Committee. These changes have been implemented with the respective License Amendments. A summary of these changes is provided in Attachment 2.

Current TS Bases pages reflecting the changes discussed in Attachments 1 and 2 are provided in Attachment 3.

If you have any questions regarding this transmittal, please contact Mrs. Candee G. Lovett at (757) 365-2178.

Very truly yours,

B. L. Stanley
Director Station Safety and Licensing
Surry Power Station

A001
NRR

Attachments:
1. Summary of TS Bases Changes Not Previously Submitted to the NRC
2. Summary of TS Bases Changes Associated with License Amendments
3. Current TS Bases Pages

Commitments made in this letter: None.

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Attachment 1
Serial No. 11-594

Summary of TS Bases Changes Not Previously Submitted to the NRC

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

SUMMARY OF TS BASES CHANGES NOT PREVIOUSLY SUBMITTED TO THE NRC

TS 3.6 Basis Revision (TS Basis Page TS 3.6-5b (superseded by Basis change associated with License Amendments 269/268 implemented on November 3, 2011))

The revision to the TS 3.6 Basis was needed as a correction to delete the sentence that stated "It [the ECST] is also sufficient to maintain one unit at hot shutdown for 2 hours, followed by a 4-hour cooldown from 547°F to 350°F (i.e., RHR operating conditions)."

This Basis change was approved on September 21, 2010 and implemented on October 20, 2011.

TS 4.1 Basis Revision (TS Basis Page TS 4.1-4)

The revision to the TS 4.1 Basis provided clarification of the surveillance test frequencies of prior to each startup if not done within the previous 30 days in TS Table 4.1-1.

This Basis change was approved on September 21, 2010 and implemented on October 20, 2011.

Attachment 2
Serial No. 11-594

Summary of TS Bases Changes Associated with License Amendments

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

SUMMARY OF TS BASES CHANGES ASSOCIATED WITH LICENSE AMENDMENTS

One-time Alternate Repair Criteria for SG Tube Repair (TS Bases Pages 3.1-14a, TS 3.1-14b, and TS 4.13-2)

This amendment established a one-time alternate repair and inspection criteria for portions of the SG tubes within the tubesheet. It was implemented on Unit 1 for the Refueling Outage 23 (Fall 2010) and the subsequent operating cycle. The Unit 2 amendment was previously implemented for the Refueling Outage 22 (Fall 2009) and the subsequent operating cycle.

The associated Bases changes were included for information in a September 30, 2009 letter (Serial No. 09-455B) and were incorporated into the Bases as part of the November 3, 2010 implementation of License Amendment 267/-- issued on November 5, 2009.

Measurement Uncertainty Recapture Power Uprate (TS Bases Pages TS 2.1-5 (superseded by Basis change associated with License Amendments 270/269 implemented on November 3, 2011) and TS 3.6-5b)

These amendments increased each unit's rate power (RP) level from 2546 megawatts thermal (MWt) to 2587 MWt and made technical specification changes necessary to support operation at the uprated power level.

The associated Bases changes were included for information in a January 27, 2010 letter (Serial No. 09-223) and were incorporated into the Bases as part of the November 3, 2010 implementation of License Amendments 269/268 issued on September 24, 2010.

Relocation of Core Operating Limits to the Core Operating Limits Report (COLR) and COLR References (TS Bases Pages TS 2.1-2, TS 2.1-3, TS 2.1-4, TS 2.3-5, TS 3.12-20, TS 4.1-5, and TS 4.1-5a)

These amendments relocated cycle-specific DNB parameters to the COLR in accordance with TSTF-339-A and revised the TS 6.2.C References to include additional methodologies.

The associated Bases changes were included for information in an October 16, 2009 letter (Serial No. 09-581) and were incorporated into the Bases as part of the November 3, 2010 implementation of License Amendments 270/269 issued on October 19, 2010.

Revised Effective Full Power Years (EFPYs) for Heatup and Cooldown Curves (TS Basis Pages TS 3.1-9 and TS 3.1-11)

These amendments revised the pressure and temperature limits curves to provide new limits that are valid to 48 EFPYs for Surry 1 and 2.

The associated Bases changes were included for information in a May 6, 2010 letter (Serial No. 10-199) and were incorporated into the Bases as part of the June 15, 2011 implementation of License Amendments 274/274 issued on May 31, 2011.

Administrative Changes to TS 3.12 and TS 6.2 (TS Basis Page TS 3.12-13)

These amendments included administrative changes that: 1) correct an error in TS 3.12.E.5, 2) delete duplicative requirements in TS 3.12.E.2 and TS 3.12.E.4, 3) relocate the shutdown margin value in TS 3.12 and the TS 3.12 Basis to the Core Operating Limits Report (COLR), and 4) expand the TS 6.2 list of parameters defined in the COLR.

The associated Basis changes were included for information in a July 12, 2010 letter (Serial No. 10-391) and were incorporated into the Bases as part of the August 26, 2011 implementation of License Amendments 275/275 issued on July 28, 2011.

Attachment 3
Serial No. 11-594

Current TS Bases Pages

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

B. In the event the Safety Limit is violated, the facility shall be placed in at least HOT SHUTDOWN within 1 hour. The safety limit is exceeded if the combination of RCS highest loop average temperature and THERMAL POWER level is at any time above the appropriate pressure line as specified in the CORE OPERATING LIMITS REPORT; or the core THERMAL POWER exceeds 118% of the rated power.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, DNB has been correlated to thermal power, reactor coolant temperature and reactor coolant pressure which are observable parameters. This correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the DNB heat flux at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB basis is as follows: there must be at least a 95% probability with 95% confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is based on the entire applicable experimental data set to meet this statistical criterion.[1]

The figure provided in the CORE OPERATING LIMITS REPORT shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation. The area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would

be required if they were based upon the design DNBR limit alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The effects of rod bowing are also considered in the DNBR analyses.

The reactor core Safety Limits are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core Safety Limits are used to define the various Reactor Protection System (RPS) functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that the variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core Safety Limits will be satisfied during steady state operation, normal operational transients, and AOOs.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 573.0°F and a steady state nominal operating pressure of 2235 psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +0.38% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state

errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also ensures that at least 99.9% of the core avoids the onset of DNB when the limiting rod is at the DNBR limit.

The fuel overpower design limit is 118% of rated power. The overpower limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

References

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1) FSAR Section 3.4

2) FSAR Section 3.3

3) FSAR Section 14.2

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3 seconds), and pressure is within the range between high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as specified in the CORE OPERATING LIMITS REPORT. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor limit is automatically reduced.[4][5]

The overpower and overtemperature protection system setpoints have been revised to include effects of fuel densification on core safety limits and to apply to 100% of design flow. The revised setpoints in the Technical Specifications will ensure that the combination of power, temperature, and pressure will not exceed the revised core safety limits as specified in the CORE OPERATING LIMITS REPORT. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips. The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The overpower protection system set points include the effects of fuel densification.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed Section 7 and specified in Section 14.2.2 of the FSAR and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

Refer to Technical Report EE-0116 for justification of the dynamic limits (time constants) for the Overtemperature ΔT and Overpower ΔT Reactor Trip functions.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 48 Effective Full Power Years (EFPY) for Units 1 and 2. The most limiting value of RT_{NDT} (222.5°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 2 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 48 EFPY for Units 1 and 2 (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 48 EFPY for Units 1 and 2 prior to a scheduled refueling outage.

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 48 EFPY for Units 1 and 2. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES - Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a main steam line break (MSLB) accident. Other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The MSLB is less limiting for site radiation releases than the SGTR. The safety analysis for the MSLB accident assumes 1 gpm total primary to secondary LEAKAGE, including 500 gpd leakage into the faulted generator. The dose consequences resulting from the MSLB and the SGTR accidents are within the limits defined in the plant licensing basis.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITIONS FOR OPERATION - RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

   No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY - In REACTOR OPERATION conditions where T_{avg} exceeds 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.1.C.5 measures leakage through each individual pressure isolation valve (PIV) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

The specified minimum water volume in the 110,000-gallon protected condensate storage tank is sufficient for 8 hours of residual heat removal following a reactor trip and loss of all offsite electrical power. If the protected condensate storage tank level is reduced to 60,000 gallons, the immediately available replenishment water in the 300,000-gallon condensate tank can be gravity-fed to the protected tank if required for residual heat removal. An alternate supply of feedwater to the auxiliary feedwater pump suctions is also available from the Fire Protection System Main in the auxiliary feedwater pump cubicle.

The five main steam code safety valves associated with each steam generator have a total combined capacity of 3,842,454 pounds per hour at their individual relieving pressure; the total combined capacity of all fifteen main steam code safety valves is 11,527,362 pounds per hour. The maximum steam flow at full power is approximately 11,444,000 pounds per hour. The combined capacity of the safety valves required by Specification 3.6 always exceeds the total steam flow corresponding to the maximum steady state power than can be obtained during three reactor coolant loop operation.

The availability of the auxiliary feedwater pumps, the protected condensate storage tank, and the main steam line safety valves adequately assures that sufficient residual heat removal capability will be available when required.

The limit on steam generator secondary side iodine-131 activity is based on limiting the inhalation dose at the site boundary following a postulated steam line break accident to a small fraction of the 10 CFR 100 limits. The accident analysis, which is performed based on the guidance of NUREG-0800 Section 15.1-5, assumes the release of the entire contents of the faulted steam generator to the atmosphere.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to COLD SHUTDOWN) are compensated for by changes in the soluble boron concentration. During POWER OPERATION, the shutdown control rod assemblies are fully withdrawn and control of power is by the control banks. A reactor trip occurring during POWER OPERATION will place the reactor into HOT SHUTDOWN. The control rod assembly insertion limits provide for achieving HOT SHUTDOWN by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted control rod assembly worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement.

The maximum shutdown margin requirement occurs at end of core life and is based on the value used in the analyses of the hypothetical steam break accident. The control rod assembly insertion limits are based on end of core life conditions. The shutdown margin for the entire cycle length shall be within the limits specified in the CORE OPERATING LIMITS REPORT. Other accident analyses with the exception of the Chemical and Volume Control System malfunction analyses are based on 1% reactivity shutdown margin. Relative positions of control banks are determined by a specified control bank overlap. This overlap is based on the consideration of axial power shape control. The specified control rod assembly insertion limits have been established to limit the potential ejected control rod assembly worth in order to account for the effects of fuel densification. The various control rod assemblies (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank; that is, with each assembly in the bank within one step (5/8 inch) of the bank position.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called the group step demand counters) and the Rod Position Indication System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one group step demand counter for each group of rods. Individual

A 2% QUADRANT POWER TILT allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod assembly and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rod assemblies producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The QPTR limit must be maintained during power operation with THERMAL POWER > 50% of RATED POWER to prevent core power distributions from exceeding the design limits.

Applicability during power operation ≤ 50% RATED POWER or when shut down is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RATED POWER or lower.

The limits of the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin. Therefore, measurement values are compared directly to the surveillance limits without applying instrument uncertainty.

The 12 hour periodic surveillance of temperature and pressure through instrument readout is sufficient to ensure that these parameters are restored to within their limits following load changes and other expected transient operation. The 12 hour surveillance of RCS total flow rate, by installed flow instrumentation, is sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions. Measurement of RCS total flow rate by performance of a precision calorimetric heat balance specified in TS Table 4.1-2A allows for the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specific surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR TRIP INSTRUMENTATION SYSTEM, and supplements to that report, WCAP-10271 Supplement 2, EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM, and supplements to that report, and WCAP-14333P, PROBABILISTIC RISK ANALYSIS OF THE RPS AND ESF TEST TIMES AND COMPLETION TIMES, as approved by the NRC and documented in SERs dated February 21, 1985, February 22, 1989, the SSER dated April 30, 1990 for WCAP-10271 and July 15, 1998 for WCAP-14333P. For those functional units not included in the generic Westinghouse probabilistic risk analyses discussed above, a plant-specific risk assessment was performed. This risk assessment demonstrates that the effect on core damage frequency and incremental change in core damage probability is negligible for the relaxations associated with the additional functional units.

Surveillance testing of instrument channels is routinely performed with the channel in the tripped condition. Only those instrument channels with hardware permanently installed that permits bypassing without lifting a lead or installing a jumper are routinely tested in the bypass condition. However, an inoperable channel may be bypassed by lifting a lead or installing a jumper to permit surveillance testing of another instrument channel of the same functional unit.

Some items in Table 4.1-1 have a test frequency of prior to each startup if not done within the previous 31 days with no applicability specified with respect to when during each startup. The following information is provided for those items to clarify when during each startup the testing is required to be performed:

- Table 4.1-1 Item 2 - Nuclear Intermediate Range - Prior to criticality if not done within the previous 31 days
- Table 4.1-1 Item 3 - Nuclear Source Range - Prior to criticality if not done within the previous 31 days
- Table 4.1-1 Item 28.A - Turbine Trip Stop Valve Closure - Prior to exceeding the P-7 setpoint if not done within the previous 31 days
- Table 4.1-1 Item 28.B - Turbine Trip Low Fluid Oil Pressure - Prior to exceeding the P-7 setpoint if not done within the previous 31 days

The refueling water storage tank is sampled weekly for Cl^- and/or F^- contaminations. Weekly sampling is adequate to detect any inleakage of contaminated water.

Main Control Room/Emergency Switchgear Room (MCR/ESGR) Envelope Isolation Actuation Instrumentation

The MCR/ESGR Envelope Isolation Actuation function provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. A functional check of the Manual Actuation function is performed every 18 months. The test frequency is based on the known reliability of the function and the redundancy available and has been shown to be acceptable through operating experience. The Surveillance Requirement will ensure that the two trains of the MCR/ESGR envelope isolation dampers close upon manual actuation of the MCR/ESGR Envelope Isolation Actuation Instrumentation and that the supply and exhaust fans in the normal ventilation system for the MCR/ESGR envelope shut down, as well as adjacent area ventilation fans. Automatic actuation of the MCR/ESGR Envelope Isolation Actuation Instrumentation is confirmed as part of the Logic Channel Testing for the Safety Injection system.

Pressurizer PORV, PORV Block Valve, and PORV Backup Air Supply

The safety-related, seismic PORV backup air supply is relied upon for two functions - mitigation of a design basis steam generator tube rupture accident and low temperature overpressure protection (LTOP) of the reactor vessel during startup and shutdown. The surveillance criteria are based upon the more limiting requirements for the backup air supply (i.e. more PORV cycles potentially required to perform the mitigation function), which are associated with the LTOP function.

The PORV backup air supply system is provided with a calibrated alarm for low air pressure. The alarm is located in the control room. Failures such as regulator drift and air leaks which result in low pressure can be easily recognized by alarm or annunciator action. A periodic quarterly verification of air pressure against the surveillance limit supplements this type of built-in surveillance. Based on experience in operation, the minimum checking frequencies set forth are deemed adequate.

RCS Flow

The frequency of 18 months for RCS flow surveillance reflects the importance of verifying flow after a refueling outage when the core has been altered, which may

have caused an alteration of the flow resistance. This surveillance requirement in Table 4.1-2A is modified by a note that allows entry into POWER OPERATION, without having performed the surveillance, and placement of the unit in the best condition for performing the surveillance. The note states that the surveillance requirement is not required to be performed until 7 days after reaching a THERMAL POWER of $\geq$ 90% of RATED POWER. The 7 day period after reaching 90% of RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. The surveillance shall be performed within 7 days after reaching 90% of RATED POWER.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in the TS 3.1.C Bases.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 24 hour frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 4.13.B

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.1.H, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG.

If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG. The surveillance is modified by a Note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The surveillance frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 4).