VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 9, 2006

U.S. Nuclear Regulatory Commission	Serial No. 06-869
Attention: Document Control Desk	SPS-LIC/CGL R0
Washington, D.C. 20555	Docket Nos. 50-280, 281
	License Nos. DPR-32, 37

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY POWER STATION UNITS 1 AND 2
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 the changes to the Bases of the Surry TS implemented since inclusion of Technical Specification 6.4.J into the Surry TS. Technical Specification 6.4.J was incorporated into the Surry TS as part of License Amendments 243/242, which were issued on July 15, 2005 and implemented on August 12, 2005.

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 Station Nuclear Safety and Operating Committee. It was determined that these changes neither required a change to the TS or license nor involved 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 with License Amendment Requests pursuant to 10CFR50.90 are also being submitted. These Bases changes were implemented with the associated License Amendments. A summary of these changes is provided in Attachment 2.

Current TS Bases pages associated with the changes discussed in Attachments 1 and 2 are provided in Attachment 3 for your information.

If you have questions regarding this submittal, please contact Mr. Gary D. Miller at (804) 273-2771.

Sincerely,

G. T. Bischof
Vice President – Nuclear Engineering

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

cc: U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW
Suite 23T85
Atlanta, Georgia 30303

Mr. N. P. Garrett
NRC Senior Resident Inspector
Surry Power Station

Commissioner
Bureau of Radiological Health
1500 East Main Street
Suite 240
Richmond, Virginia 23218

Mr. S. P. Lingam
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Mail Stop 8G9A
Rockville, Maryland 20852

Mr. S. R. Monarque
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Mail Stop 8-H12
Rockville, Maryland 20852

Attachment 1

Summary of TS Bases Changes Not Previously Submitted to the NRC

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

SUMMARY OF TS BASES CHANGES NOT PREVIOUSLY SUBMITTED TO THE NRC

TS 3.16 Basis Revision Addressing Engineered Safeguards Equipment Powered from an Emergency Bus (TS Basis Page TS 3.16-5)

This TS 3.16 Basis revision modifies the list of Engineered Safeguards Equipment to delete the containment vacuum pump, to change the number of control area air conditioning subcomponents to reflect the current plant configuration, and to revise the auxiliary feedwater pump description.

This Basis revision was approved on April 21, 2006.

TS 3.1.E Basis Revision Regarding Moderator Temperature Coefficient (TS Basis Page TS 3.1-18)

This TS 3.1.E Basis revision states that the moderator temperature coefficient will be most positive near the beginning of cycle life. This revision is appropriate to reflect implementation of the Westinghouse integral fuel burnable absorber (IFBA) product and is valid for both non-IFBA and IFBA fuel.

This Basis revision was approved on March 24, 2006.

Attachment 2

Summary of TS Bases Changes
Associated with License Amendments

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

SUMMARY OF TS BASES CHANGES ASSOCIATED WITH LICENSE AMENDMENTS

Deletion of TS 4.0.5, Creation of Inservice Testing Program, and Addition of TS Bases Control Program (TS Bases Pages TS 4.0-2, TS 4.0-3, TS 4.0-4, TS 4.0-5, TS 4.2-2, TS 4.8-3, and TS 4.11-4)

This change deleted the Inservice Inspection (ISI) and Inservice Testing (IST) Requirements in TS 4.0.5, relocated the IST requirements to the administrative section of the TS as a program, and added a TS Bases Control Program to the administrative section of the TS.

The associated Bases changes were included for information in a November 4, 2004 letter (Serial No. 04-666) and were incorporated into the Bases upon approval of the associated TS by License Amendments 243/242 issued on July 15, 2005.

Revision of Auxiliary Feedwater Requirements (TS Bases Pages TS 3.6-5, TS 3.6-5a, TS 3.6-5b, TS 3.6-6, TS 3.9-2, TS 3.16-6, TS 4.8-3, and TS 4.8-4)

This change revised the auxiliary feedwater (AFW) requirements to eliminate the inconsistency between the AFW pump requirements and the required actions, establish consistency with the Improved TS, and add an AFW flowpath allowed outage time along with required actions.

The associated Bases changes were included for information in a March 8, 2005 letter (Serial No. 05-107) and were incorporated into the Bases upon approval of the associated TS by License Amendments 246/245 issued on February 23, 2006.

Revision of Pressure-Temperature Limits and Low-Temperature Overpressure Protection System Values (TS Basis Pages TS 3.1-9, TS 3.1-10 and TS 3.1-11)

This change revised the reactor coolant pressure and temperature operating limits, low-temperature overpressure protection system (LTOPS) setpoint value, and LTOPS enable temperature basis. This change superseded the revised limits included in previous License Amendments 245/244 and returned the limits and values to those in place prior to the issuance of License Amendments 245/244.

The associated Basis changes were included for information in an April 20, 2006 letter (Serial No. 06-335) and were incorporated into the Basis upon approval of the associated TS by License Amendments 248/247 issued on June 29, 2006.

Revision of Accident Monitoring Instrumentation Requirements (TS Bases Pages TS 3.7-7, TS 3.7-8, TS 3.7-9, and TS 4.1-3)

This change revised the accident monitoring instrumentation listing, allowed outage times, requirements, and surveillances to be consistent with the requirements of the Improved TS for post-accident monitoring instrumentation. In addition, editorial changes were made in the Bases for TS 3.7 and 4.1.

The associated Bases changes were included for information in a July 21, 2005 letter (Serial No. 05-422) and were incorporated into the Bases upon approval of the associated TS by License Amendments 247/246 issued on May 31, 2006.

Attachment 3

Current TS Bases Pages

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

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 28.8 Effective Full Power Years (EFPY) and 29.4 EFPY for Units 1 and 2, respectively. The most limiting value of RT_{NDT} (228.4°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 1 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 28.8 EFPY and 29.4 EFPY for Units 1 and 2, respectively (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 28.8 EFPY or 29.4 EFPY for Units 1 and 2, respectively, prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T - RT_{NDT} + 160)] \qquad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

where, K_{IM} is the stress intensity factor caused by membrance (pressure) stress.

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 28.8 EFPY and 29.4 EFPY for Units 1 and 2, respectively. 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.

E. Minimum Temperature for Criticality

Specifications

1. Except during LOW POWER PHYSICS TESTS, the reactor shall not be made critical at any Reactor Coolant System temperature above which the moderator temperature coefficient is more positive than the limit specified in the CORE OPERATING LIMITS REPORT. The maximum upper limit for the moderator temperature coefficient shall be:

   a. + 6 pcm/°F at less than 50% of RATED POWER, or

   b. + 6 pcm/°F at 50% of RATED POWER and linearly decreasing to 0 pcm/°F at RATED POWER.

2. In no case shall the reactor be made critical with the Reactor Coolant System temperature below the limiting value of RT_{NDT} + 10°F, where the limiting value of RT_{NDT} is as determined in Part B of this specification.

3. When the Reactor Coolant System temperature is below the minimum temperature as specified in E-2 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to primary coolant depressurization.

4. The reactor shall not be made critical when the Reactor Coolant System temperature is below 522°F.

Basis

During the early part of a fuel cycle, the moderator temperature coefficient may be calculated to be slightly positive at coolant temperatures in the power operating range. The moderator temperature coefficient will be most positive near the beginning of cycle life, generally corresponding to when the boron concentration in the coolant is the greatest. Later in the cycle, the boron concentration in the coolant will generally be lower and the moderator temperature coefficient will be less positive or will be negative in the power operating range. At the beginning of cycle life, during pre-operational physics tests, measurements are made to determine that the moderator temperature coefficient is less than the limit specified in the CORE OPERATING LIMITS REPORT.

Basis

A reactor which has been shutdown from power requires removal of core residual heat. While reactor coolant temperature or pressure is > 350°F or 450 psig, respectively, residual heat removal requirements are normally satisfied by steam bypass to the condenser. If the condenser is unavailable, steam can be released to the atmosphere through the safety valves or power operated relief valves. The capability to supply feedwater to the generators is normally provided by the operation of the Condensate and Feedwater Systems.

The Auxiliary Feedwater System provides a source of feedwater to the secondary side of the steam generators at times when the Feedwater System is not available, thereby maintaining heat sink capabilities of the steam generators. The Auxiliary Feedwater System provides heat removal until normal feedwater flow is restored or until an orderly cooldown to Reactor Coolant System conditions where the Residual Heat Removal System can be placed in service. The Auxiliary Feedwater System for each unit consists of two motor driven pumps, one turbine driven pump, a 110,000 gallon protected condensate storage tank, and associated common piping, redundant headers, valves, controls, and instrumentation. Although the flowpaths from the pumps to the steam generators include common piping, the configuration of the system provides two redundant flowpaths. The components in one flowpath are supplied by the H emergency bus, while the other is supplied by the J emergency bus. The auxiliary feedwater design basis accident is a loss of normal feedwater with offsite power available (the reactor coolant pumps running). The auxiliary feedwater flow required to remove the heat and cool the unit to residual heat removal conditions for this design basis case can be provided by any combination of two auxiliary feedwater pumps.

Refer to the Basis of Specification 4.8 for a discussion of auxiliary feedwater pump operability considerations.

Regarding the allowed outage times for auxiliary feedwater pump inoperability, Specification 3.6.E allows 7 days versus a 72 hour allowed outage time in Specification 3.6.F.1. The longer allowed outage time is based on the reduced decay heat following refueling and prior to reactor criticality.

In the unlikely event of loss of auxiliary feedwater capability on the affected unit (i.e., with all required auxiliary feedwater pumps inoperable or with both redundant flowpaths having an inoperable component or instrumentation), the required action is to immediately initiate action to restore operability of one inoperable pump or of the inoperable component or instrumentation in one flowpath. With such a loss of auxiliary feedwater capability, the unit is in a seriously degraded condition. In this condition, the unit should not be perturbed by any action, including a power change, which could result in a plant transient or trip. The seriousness of this condition requires that action be taken immediately to restore operability, where immediately means the required action should be pursued without delay and in a controlled manner. Under these circumstances, Specification 3.0.1 and all other required actions directing mode changes are suspended until one inoperable pump or the inoperable component or instrumentation in one flowpath is restored to operable status, because taking those actions could place the unit in a less safe condition.

In the event of complete loss of electrical power to the station, residual heat removal would continue to be assured by the availability of either the turbine driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps and the 110,000-gallon protected condensate storage tank.

In the event of a fire or high energy line break which would render the auxiliary feedwater pumps inoperable on the affected unit, residual heat removal would continue to be assured by the availability of either the turbine driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps from the opposite unit. A minimum of two auxiliary feedwater pumps are required to be operable* on the opposite unit to ensure compliance with the design basis accident analysis assumptions, in that auxiliary feedwater can be delivered via the cross-connect, even if a single active failure results in the loss of one of the two pumps. In addition, the requirement for operability of the opposite unit's emergency power system is to ensure that auxiliary feedwater from the opposite unit can be supplied via the cross-connect in the event of a common-mode failure of all auxiliary feedwater pumps in the affected unit due to a high energy line break in the main steam valve house. Without this requirement, a single failure (such as loss of the shared backup diesel generator) could result in loss of power to the opposite unit's emergency buses in the event of a loss of offsite power, thereby rendering the cross-connect inoperable. The longer allowed outage time for the opposite unit's emergency power system is based on the low probability of a high energy line break in the main steam valve house coincident with a loss of offsite power.

* excluding automatic initiation instrumentation

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. It 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). 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 nominal power rating steam flow is 11,260,000 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.

REFERENCES

UFSAR Section 4, Reactor Coolant System

UFSAR Section 9.3, Residual Heat Removal System

UFSAR Section 10.3.1, Main Steam System

UFSAR Section 10.3.2, Auxiliary Steam System

UFSAR Section 10.3.5, Condensate and Feedwater Systems

UFSAR Section 10.3.8, Secondary Vent and Drain Systems

UFSAR Section 14.2.11, Loss of Normal Feedwater

UFSAR Section 14.3.2, Rupture of a Main Steam Pipe

UFSAR Appendix 14B, Effects of Piping System Breaks Outside Containment steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break.[(3)]

Accident Monitoring Instrumentation

The primary purpose of accident monitoring instrumentation is to display unit parameters that provide information required by the control room operators during and following accident conditions. In response to NUREG-0737 and Regulatory Guide (RG) 1.97, Revision 3, a programmatic approach was developed in defining the RG 1.97-required equipment for Surry. The Surry RG 1.97 program review examined existing instrumentation with respect to the RG 1.97 design and qualification requirements. The operability of RG 1.97 instrumentation ensures that sufficient information is available on selected unit parameters to monitor and assess unit status and response during and following an accident. The availability of accident monitoring instrumentation is important so that the consequences of corrective actions can be observed and the need for and magnitude of further actions can be determined.

RG 1.97 applied a graded approach to post-accident indication by using a matrix of variable types versus variable categories. RG 1.97 delineates design and qualification criteria for the instrumentation used to measure five variable types (Types A, B, C, D, and E). These criteria are divided into three separate categories (Categories 1, 2, and 3), providing a graded approach that depended on the importance to safety of the measurement of a specific variable. Category 1 variables, listed in Table 3.7-6, are defined as follows:

Category 1 - are the key variables deemed risk significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions,
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release, and
- Provide information regarding the release of radioactive materials to allow early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

The RG 1.97 criteria on redundancy requirements apply to Category 1 variables only and address single-failure criteria and supporting features, including power sources. Failures of the instrumentation, its supporting features, and/or its power source resulting in less than the required number of channels necessitate entry into the required actions.

The 30 day allowed outage time applies when one (or more) function(s) in Table 3.7-6 has one required channel that is inoperable. The 30 day allowed outage time to restore one inoperable required channel to OPERABLE status is appropriate considering the remaining channel is OPERABLE, the passive nature of the instrument (i.e., no automatic action is assumed to occur from this instrumentation), and the low probability of an event requiring accident monitoring instrumentation during this interval. The 7 day allowed outage time applies when one (or more) function(s) in Table 3.7-6 has two required channels that are inoperable. The 7 day allowed outage time to restore one of the two inoperable required channels to OPERABLE status is appropriate based on providing a reasonable time for the repair and the low probability of an event requiring accident monitoring instrument operation. Long-term operation with two required channels inoperable in a function and with an alternate indication is not acceptable because the alternate indication may not fully meet the performance qualification requirements applied to the accident monitoring instrumentation. Requiring restoration of one of the two inoperable channels limits the risk that the accident monitoring instrumentation function could be in a degraded condition should an accident occur. If there is no preplanned alternate, the 7 day allowed outage time is followed by a requirement to be in HOT SHUTDOWN within the next 6 hours and be less than 350°F and 450 psig within the following 12 hours. If the 30 day allowed outage time or 7 day allowed outage time to restore an inoperable channel to OPERABLE status is exceeded and either a redundant channel or a preplanned alternate method of monitoring is OPERABLE, a report to the NRC within the next 14 days is required. The report to the NRC in lieu of a shutdown is appropriate because the instrument functional capability has not been lost and given the low likelihood of unit conditions that would require the information provided by the accident monitoring instrumentation.

Note that the Categories 2 and 3 RG 1.97 variables are addressed in a licensee controlled document and are defined as follows:

> Category 2 - provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status.
>
> Category 3 - is the least stringent and is applied to backup and diagnostic instrumentation.

Explosive Gas Monitoring

Instrumentation is provided for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the Waste Gas Holdup System. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

Non-Essential Service Water Isolation System

The operability of this functional system ensures that adequate intake canal inventory can be maintained by the Emergency Service Water Pumps. Adequate intake canal inventory provides design service water flow to the recirculation spray heat exchangers and other essential loads (e.g., control room area chillers, charging pump lube oil coolers) following a design basis loss of coolant accident with a coincident loss of offsite power. This system is common to both units in that each of the two trains will actuate equipment on each unit.

Clarification of Operator Actions

The Operator Actions associated with Functional Units 10 and 16 on Table 3.7-1 require the unit to be reduced in power to less than the P-7 setpoint (10%) if the required conditions cannot be satisfied for either the P-8 or P-7 permissible bypass conditions. The requirement to reduce power below P-7 for a P-8 permissible bypass condition is necessary to ensure consistency with the out of service and shutdown action times assumed in the WCAP-10271 and WCAP-14333P risk analyses by eliminating the potential for a scenario that would allow sequential entry into the Operator Actions (i.e., initial entry into the Operator Action with a reduction in power to below P-8, followed by a second entry into the Operator Action with a reduction in power to below P-7). This scenario would permit sequential allowed outage time periods that may result in an additional 72 hours that was not assumed in the risk analysis to place a channel in trip or to place the unit in a condition where the protective function was not necessary.

References

(1) UFSAR - Section 7.5

(2) UFSAR - Section 14.5

(3) UFSAR - Section 14.3.2

7. Two emergency diesel generators OPERABLE as explained in Section 3.16.

B. The requirements of Specification 3.9-A items 3, 4, 5, 6, and 7 may be modified as provided in Section 3.16-B.

Basis

During startup of a unit, the station's 4,160V and 480V normal and emergency buses are energized from the station's 34.5KV buses. At reactor power levels greater than 5 percent of rated power the 34.5KV buses are required to energize only the emergency buses because at this power level the station generator can supply sufficient power to the normal 4,160V and 480V lines to operate the unit. Three reactor coolant loop operation with all 4,160V and 480V buses energized is the normal mode of operation for a unit.

The electrical power requirements and the emergency power testing requirements for the auxiliary feedwater cross-connect are contained in TS 3.6.C.4.c and TS 4.6, respectively.

References

FSAR Section 8.4 Station Service Systems

FSAR Section 8.5 Emergency Power Systems

The diesel generators function as an on-site back-up system to supply the emergency buses. Each emergency bus provides power to the following operating Engineered Safeguards equipment:

A. One containment spray pump

B. One charging pump

C. One low head safety injection pump

D. One recirculation spray pump inside containment

E. One recirculation spray pump outside containment

F. One motor-driven auxiliary feedwater pump

G. One motor control center for valves, instruments, control air compressor, fuel oil pumps, etc.

H. Control area air conditioning equipment - two air recirculating units, one water chilling unit, one service water pump, and one chilled water circulating pump

I. One charging pump service water pump

The day tanks are filled by transferring fuel from any one of two buried tornado missile protected fuel oil storage tanks, each of 20,000 gal capacity. Two of 100 percent capacity fuel oil transfer pumps per diesel generator are powered from the emergency buses to assure that an operating diesel generator has a continuous supply of fuel. The buried fuel oil storage tanks contain a seven (7) day supply of fuel, 35,000 gal minimum, for the full load operation of one diesel generator; in addition, there is an above ground fuel oil storage tank on-site with a capacity of 210,000 gal which is used for transferring fuel to the buried tanks.

One of the two buried fuel oil storage tanks may be inoperable to permit inspection and related repair of that buried fuel oil storage tank. While one tank is removed from service, the remaining buried fuel oil storage tank supplies fuel oil to the EDGs of both units. Prior to removal of one buried tank from service and while it is inoperable, verification of the volume in the remaining buried fuel oil storage tank and the above ground fuel oil storage tank is required to ensure an adequate source of fuel oil remains available onsite. In addition, verification of the offsite replacement fuel oil supply is also required. While one buried tank is out of service, the verification of the onsite and offsite fuel oil sources continues to support full load operation of one diesel generator for seven days.

If a loss of normal power is not accompanied by a loss-of-coolant accident, the safeguards equipment will not be required. Under this condition the following additional auxiliary equipment may be operated from each emergency bus:

A. One component cooling pump

B. One residual heat removal pump

C. One motor-driven auxiliary steam generator feedwater pump

The emergency buses in each unit are capable of being interconnected under strict administrative procedures so that the equipment which would normally be operated by one of the diesels could be operated by the other diesel, if required.

The electrical power requirements and the emergency power testing requirements for the auxiliary feedwater cross-connect are contained in TS 3.6.C.4.c and TS 4.6 respectively.

BASES

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during all operating conditions for which the Limiting Conditions for Operation are applicable.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

4.0.3 This specification establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the operability requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when surveillance requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the surveillance requirements. This specification also clarifies that the Action Statement requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the Action Statement requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the surveillance requirement within the allowable outage time limits of the Action Statement requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the operability requirements of a Limiting Condition for Operation. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B), unless it meets an exception listed therein, because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the Action Statement requirements are less than 24 hours or a shutdown is required to comply with Action Statement requirements, e.g., Specification 3.0.1, a 24 hour allowance is provided to permit a delay in implementing the Action Statement requirements. This provides an adequate time limit to complete surveillance requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with Action Statement requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of surveillance requirements that become applicable as a consequence of condition changes imposed by Action Statement requirements and for completing surveillance requirements that are applicable when an exception to the requirements of Specifications 4.0.4 is allowed. If a surveillance is not completed within the 24 hour allowance, the time limits of the Action Statement requirements are applicable at that time. When a surveillance is performed within the 24 hour allowance and the surveillance requirements are not met, the time limits of the Action Statement requirements are applicable at the time that the surveillance is terminated.

Surveillance requirements do not have to be performed on inoperable equipment because the Action Statement requirements define the remedial measures that apply. However, the surveillance requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

4.0.4 This specification establishes the requirement that all applicable surveillances must be met before entry into an operational condition specified in the applicability statement. The purpose of this specification is to ensure that system and component operability requirements or parameter limits are met before entry into a condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in operational conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable surveillance requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

Exceptions to Specification 4.0.4 allow performance of surveillance requirements associated with a Limiting Condition for Operation after entry into the applicable operational condition.

When a shutdown is required to comply with Action Statement requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower condition of operation.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process systems instrumentation errors resulting from drift within the individual instruments are normally negligible.

During the interval between periodic channel tests and daily check of each channel, a comparison between redundant channels will reveal any abnormal condition resulting from a calibration shift, due to instrument drift of a single channel.

During the periodic channel test, if it is deemed necessary, the channel may be tuned to compensate for the calibration shift. However, it is not expected that this will be required at any fixed or frequent interval.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (power level) channels, and once per 18 months for the process system channels are considered acceptable.

Testing

The OPERABILITY of the Reactor Trip System and ESFAS instrumentation systems and interlocks ensures that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy are maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the RTS and ESFAS instrumentation, and 3) sufficient system functional capability is available from diverse parameters.

Sensitized stainless steel augmented inspections were added to assure piping integrity of this classification.

Items 2.1.1-2.1.3

The examinations required by these items utilize the periodically updated ASME Section XI Boiler and Pressure Vessel Code for the augmented examinations. The surface and volumetric examinations required by items 2.1.1 and 2.1.2 will be conducted at three times the frequency required by the Code in an interval. In addition to the Code required pressure testing, visual examinations will be conducted, while the piping is pressurized by the procedures defined in Tables 4.1-3A & B of Technical Specification 4.1, concerning flushing of sensitized stainless steel piping. Weld selection criteria are modified from the Code for Class 1 welds, since stress level information as correlated to weld location is unavailable for Surry.

Item 2.2.1

The sensitized stainless steel located in the containment and recirculation spray rings in the overhead of containment are classified ASME Class 2 components. These components are currently exempted by ASME Section XI from surface and volumetric examination requirements. As such, an augmented program will remain in place requiring visual (VT-1) examination of these components for evidence of cracking. Additionally, sections of the piping will be examined by liquid penetrant inspection when the piping is visually inspected.

Basis

The correct alignment for manual, power operated, and automatic valves in the Auxiliary Feedwater System steam and water flowpaths, including the cross-connect flowpath, will provide assurance that the proper flowpaths exist for system operation. This position check does not include: 1) valves that are locked, sealed or otherwise secured in position since they are verified to be in their correct position prior to locking, sealing or otherwise securing; 2) vent, drain or relief valves on those flowpaths; and, 3) those valves that cannot be inadvertently misaligned such as check valves. This surveillance does not require any testing or valve manipulation. It involves verification that those valves capable of being mispositioned are in the correct position.

Valves in the auxiliary feedwater flowpaths to the steam generators and cross-connect flow path are tested periodically in accordance with the Inservice Testing Program. The auxiliary feedwater pumps are tested periodically in accordance with the Inservice Testing Program to demonstrate operability. Verification of the developed head of each auxiliary feedwater pump ensures that the pump performance has not degraded. Flow and differential head tests are normal inservice testing requirements. Because it is sometimes undesirable to introduce cold auxiliary feedwater into the steam generators while they are operating, the inservice testing is typically performed on recirculation flow to the 110,000 gallon Emergency Condensate Storage Tank.

Appropriate surveillance and post-maintenance testing is required to declare equipment OPERABLE. Testing may not be possible in the applicable plant conditions due to the necessary unit parameters not having been established. In this situation, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible, and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a condition where other necessary surveillance or post maintenance tests can be completed. Relative to the turbine driven auxiliary feedwater pump, Specification 4.8.A.3.a is modified by a note indicating that the developed head test of the turbine driven pump should be deferred until suitable conditions are established; this deferral is required because there may be insufficient steam pressure to perform the test.

The auxiliary feedwater pumps are capable of supplying feedwater to the opposite unit's steam generators. For a main steam line break or fire event in the Main Steam Valve House, one of the opposite units auxiliary feedwater pumps is required to supply feedwater to mitigate the consequences of those accidents. Therefore, when considering a single failure, both motor driven auxiliary feedwater pumps are required to be OPERABLE* during shutdown to support the opposite unit if the Reactor Coolant System temperature or pressure of the opposite unit is greater than 350°F and 450 psig, respectively. Thus, to establish operability* the motor driven auxiliary feedwater pumps will continue to be tested quarterly on the same STAGGERED TEST BASIS when the unit is shutdown to support the opposite unit.

The capacity of the Emergency Condensate Storage Tank and the flow rate of any one of the three auxiliary feedwater pumps in conjunction with the water inventory of the steam generators is capable of maintaining the plant in a safe condition and sufficient to cool the unit down.

Proper functioning of the steam turbine admission valve and the ability of the auxiliary feedwater pumps to start will demonstrate the integrity of the system. Verification of correct operation can be made both from instrumentation within the Main Control Room and direct visual observation of the pumps.

---

* excluding automatic initiation instrumentation

References

UFSAR Section 10.3.1, Main Steam System

UFSAR Section 10.3.2, Auxiliary Steam System

UFSAR Section 10.3.5, Condensate and Feedwater Systems

The system tests demonstrate proper automatic operation of the Safety Injection System. A test signal is applied to initiate automatic operation action and verification is made that the components receive the safety injection signal in the proper sequence. The test may be performed with the pumps blocked from starting. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.

During reactor operation, the instrumentation which is depended on to initiate safety injection is checked periodically, and the initiating circuits are tested in accordance with Specification 4.1. In addition, the active components (pumps and valves) are to be periodically tested to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval is determined in accordance with the Inservice Testing Program. The accumulators are a passive safeguard.

References

UFSAR Section 6.2, Safety Injection System