VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 31, 2005

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

Serial No. 05-319
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
TECHNICAL SPECIFICATIONS BASES CHANGES
REVISION OF REFERENCES FOR OFFSITE DOSE LIMITS

Virginia Electric and Power Company (Dominion) implemented the Alternative Source Term (AST) as the design basis for Surry Power Station by TS Amendments 230/230 issued on March 8, 2002. The approval was based on reanalysis of the Loss of Coolant Accident and the Fuel Handling Accident. Subsequently, the Locked Rotor, Main Steam Line Break, and Steam Generator Tube Rupture accidents were also reanalyzed with the AST methodology as a follow on to the previously approved AST analyses.

Depending on the accident being analyzed, the offsite dose limits associated with the AST are found in 10CFR50.67, "Accident source term" and/or Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The AST offsite dose limits of 10CFR50.67 supercede the offsite dose limits of 10CFR100 for those applications that fully implement the AST. However, certain Surry TS Bases still included references to the superceded 10CFR100 offsite dose limits rather that the AST offsite dose limits contained in 10CFR50.67 and/or Regulatory Guide 1.183.

Consequently, Dominion has approved changes to the following Technical Specifications (TS) Bases to revise the references for the offsite dose limits:

- TS 3.1 – Reactor Coolant System
- TS 3.6 – Turbine Cycle
- TS 3.8 – Containment
- TS 4.12 – Auxiliary Ventilation Exhaust Filter Trains

The revised TS Bases pages are provided in the attachment. The TS Bases revisions were reviewed and approved by the Station Nuclear Safety and Operating Committee. The 10CFR50.59 review of the TS Bases revisions concluded that NRC approval prior

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to implementation was not required. Therefore, the revisions to the TS Bases were incorporated on December 30, 2004.

A UFSAR change request has also been prepared to revise the references for the offsite dose limits, consistent with these TS Bases revisions, and will be included in the next scheduled UFSAR update in accordance with 10CFR50.71(e).

The TS Bases revisions are being provided for your information. Please note that TS Bases revisions that are implemented as part of TS change requests have been provided to the NRC for information with their associated TS change request submittals.

Should you have any questions regarding these TS Bases changes, please contact Mr. Gary D. Miller at (804) 273-2771.

Very truly yours,

E. S. Grecheck
Vice President – Nuclear Support Services

Attachment

<u>Commitment made in this letter:</u>

1. A UFSAR change request has also been prepared to revise the references for the offsite dose limits, consistent with the TS Bases revisions, and will be included in the next scheduled UFSAR update in accordance with 10 CFR 50.71(e).

cc: U. S. Nuclear Regulatory Commission
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ATTACHMENT

Revised Technical Specification Bases Pages

TS 3.1 – Reactor Coolant System
TS 3.6 – Turbine Cycle
TS 3.8 – Containment
TS 4.12 – Auxiliary Ventilation Exhaust Filter Trains

Attachment

Revised Technical Specification Bases Pages
Surry Power Station Units 1 and 2

Summary of Change:

This change to the Technical Specifications Bases is being made to change the references to regulatory offsite dose limits from 10 CFR 100 to 10 CFR 50.67 and/or RG 1.183 to reflect the current design and licensing bases based upon the implementation of the Alternative Source Term.

DELETE	DATED	SUBSTITUTE	DATED
TS 3.1-17	08-03-95	TS 3.1-17	11-08-04
TS 3.6-5a	06-07-99	TS 3.6-5a	11-08-04
TS 3.8-3	09-07-93	TS 3.8-3	11-08-04
TS 4.12-6	05-14-01	TS 4.12-6	11-08-04

Permitting startup and/or REACTOR OPERATION to continue for limited time periods with the reactor coolant's specific activity > 1.0 μCi/cc but < 10.0 μCi/cc DOSE EQUIVALENT I-131 accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER. Although the analysis of a steam generator tube rupture initiated with primary coolant activity at the 10.0 μCi/cc transient limit shows offsite doses less than or equal to the Regulatory Guide (RG) 1.183 limits, operation at the transient limit is restricted to no more than 10 percent of the unit's yearly operating time to limit the risk of appreciable release following a postulated steam generator tube rupture.

The basis for the 500°F temperature contained in the Specification is that the saturation pressure corresponding to 500°F, i.e., 680.8 psia, is well below the pressure at which the atmospheric relief valves on the secondary side could be actuated.

The accident analysis examines two cases of iodine spiking. For the case with a pre-existing iodine spike, the transient coolant activity limit of 10.0 μCi/cc is assumed. For the case of a concurrent spike, the initial activity is assumed to correspond to the steady state limit of 1.0 μCi/cc. The concurrent iodine spike is modeled with a conservative iodine appearance rate. Both cases show doses at the exclusion area and low population zone boundaries which are less than or equal to the RG 1.183 limits and control room doses which are within the General Design Criterion (GDC) 19 guidelines.

Measurement of $\bar{E}$ will be performed at least twice annually. Calculations required to determine $\bar{E}$ will consist of the following:

1. $\bar{E}$ shall be the average (weighed in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes.

3. A calculation of $\bar{E}$ by appropriate weighing of each nuclide's beta and gamma energy with its concentration as determined in (1) above.

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 dose at the site boundary following a postulated steam line break accident to the Regulatory Guide 1.183 limits. The accident analysis assumes the release of the entire contents of the faulted steam generator to the atmosphere.

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Otherwise, place the unit in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

D. Internal Pressure

1. Containment air partial pressure shall be maintained within the acceptable operation range as identified in Figure 3.8-1 whenever the Reactor Coolant System temperature and pressure exceed 350°F and 450 psig, respectively.

   a. With the containment air partial pressure outside the acceptable operation range, restore the air partial pressure to within acceptable limits within 1 hour or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Basis

CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment will be restricted to those leakage paths and associated leak rates assumed in the accident analysis. These restrictions, in conjunction with the allowed leakage, will limit the site boundary radiation dose to the applicable limits of 10 CFR 50.67 or Regulatory Guide 1.183 during accident conditions.

The operability of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. The opening of manual or deactivated automatic containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and

A pressure drop across the combined HEPA filters and charcoal adsorbers of less than 7 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Operation of the filtration system for a minimum of 15 minutes a month prevents moisture buildup in the filters and adsorbers.

The frequency of tests and sample analysis of the degradable components of the system, i.e., the HEPA filter and charcoal adsorbers, is based on actual hours of operation to ensure that they perform as evaluated. System flow rates and air distribution do not change unless the ventilation system is radically altered.

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

The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99.5 percent removal of DOP particulates. The heat release from operating ECCS equipment limits the relative humidity of the exhaust air to less than 80 percent even when outdoor air is assumed to be 100 percent relative humidity and all ECCS leakage evaporates into the exhaust air stream. Methyl iodide testing to a penetration less than or equal to 14 percent (applying a safety factor of 2) demonstrates the assumed accident analysis efficiencies of 70 percent for methyl iodide and 90 percent for elemental iodine. This conclusion is supported by a July 10, 2000 letter from NCS Corporation that stated "Nuclear grade activated carbon, when tested in accordance with ASTM D3803-1989 (methyl iodide...) to a penetration of 15%, is more conservative than testing the same carbon in accordance with ASTM D3803-1979 (elemental iodine...) to a penetration of 5%. ...As a general rule, you may expect the radioiodine penetration through nuclear grade activated carbon to increase from 20 to 100 times when switching from elemental iodine to methyl iodide testing." Therefore, the efficiencies of the HEPA filters and charcoal adsorbers are demonstrated to be as specified, at flow rates, temperatures, velocities, and relative humidities which are less than the design values of the system, the resulting doses will be less than or equal to the limits specified in 10 CFR 50.67 or Regulatory Guide 1.183 for the accidents analyzed. The demonstration of bypass 1% and demonstration of 86 percent methyl iodide removal efficiency will assure the required capability of the adsorbers is met or exceeded.