VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 6, 2001

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

Serial No.	00-303B
NLOS/ETS-CGL	R2
Docket Nos.	50-280
	50-281
License Nos.	DPR-32
	DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS AND BASES CHANGE -
RPS AND ESFAS ANALOG INSTRUMENTATION SURVEILLANCE FREQUENCY
CHANGE FROM MONTHLY TO QUARTERLY
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

In a June 16, 2000 letter (Serial No. 00-303), as supplemented on September 27, 2000 (Serial No. 00-303A), Virginia Electric and Power Company (Dominion) requested amendments, in the form of revisions to the Technical Specifications (TS) to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2. The proposed amendments would revise TS 3.7 and TS Tables 3.7-1, 3.7-2, 3.7-3, and 4.1-1. The proposed changes: a) revise the surveillance frequency for Reactor Protection System and Engineered Safety Features Actuation System analog channels from monthly to quarterly, b) decrease the frequency for most permissives to a refueling interval, c) increase the time allowed to perform maintenance on an inoperable instrument channel, and d) revise associated action statements consistent with NUREG-1431.

During November 18 and December 13, 2000 telephone conference calls to discuss the proposed changes, the bases for the allowed outage times for specific functional units were discussed. At that time, it was noted that the generic risk analysis performed in support of the underlying reference documents (WCAPs 10271 and 14333 and the associated SERs) for the extended allowed outage and channel bypass times did not adequately support the extension of several of the allowed outage times included in the proposed Technical Specifications. In addition, the staff requested additional supporting information to establish the basis of a revised action statement and discuss the elimination of several surveillance requirements for miscellaneous nonsafety-related instruments from Technical Specifications. The additional information is provided in Attachment 1 to this letter.

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In order to extend the allowed outage times for the functional units that were not included in the generic analysis, we have completed a plant specific risk assessment to establish a basis for the extended allowed outage and channel bypass times. A discussion of the plant-specific risk assessment and a revised Technical Specification Basis page TS 4.1-4 reflecting the plant-specific risk assessment are provided in Attachment 2 to this letter. Please replace the revised basis page included in Attachment 2 to complete your review of our June 16, 2000 submittal.

Should you have any questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President - Nuclear Engineering

Attachments:

1. Response to Request for Additional Information
2. Discussion of and Revised TS 4.1 Basis

Commitments made in this letter: None

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

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

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

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 6th day of June, 2001.

My Commission Expires: 3-31-04.

Maggie McClure
Notary Public

(SEAL)

Attachment 1
Response to Request for Additional Information for
Proposed Technical Specifications and Bases Change - RPS and ESFAS Analog Instrumentation Surveillance Frequency Change

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

NRC Question 1

In what licensee-controlled document or program are the items being removed from Table 4.1-1 going to reside?

Dominion Response

Channel Descriptions 12, 13, 14, 16, 18, 21, 24, 25, 27, 29, and 31 are being proposed for deletion from Table 4.1-1. These items are surveillance requirements for instruments that do not have an associated Limiting Condition for Operation nor perform a specific safety function and, thus, do not require relocation to either the UFSAR or Technical Requirements Manual. The surveillance requirements (calibration and test) for these items will be incorporated into procedures, for which procedural control processes are in place.

NRC Question 2

What is the basis for not performing the setpoint verification of functional units 8, 32b, 33a, and 33b on Table 4.1?

Dominion Response

The note regarding setpoint verification on functional units 8, 32.b, 33.a, and 33.b in Table 4.1 was being proposed as clarification to the existing surveillance requirement for the relays. Due to the channel design the sensing relay setpoints are not currently checked at power.

- The functional test for 4 KV voltage and frequency (functional unit 8) uses installed test switches to confirm proper relay operation. Relay energization is confirmed by illumination of the control room annunicators. Setpoint verification would require the installation of a jumper(s).

- The functional test and calibration (including setpoint verification) of the RCP undervoltage start of the steam driven AFW pump (functional unit 32.b) are not required at power. The functional test is required to be performed on an 18 month frequency and within 31 days prior to each startup and is conducted with the plant in a shutdown condition. Setpoint verification is not performed as part of the functional test.

- The functional test for the undervoltage (functional unit 33.a) and degraded voltage (functional unit 33.b) verify the proper operation of the instrument channel and the associated actuation logic relays by using knife switches to energize and de-energize the relays while verifying indicating lights change state appropriately. Although the existing design would permit verifying the setpoint without lifting leads or installing jumpers, the relay paddle plug would have to be removed from the circuit to de-energize the relay and replaced with a relay test plug to permit setpoint verification of the relays at power.

The refueling calibration history for the relays associated with functional units 8, 32.b, 33.a, and 33.b indicates that the relays' setpoints are very repeatable and maintain calibration within their channel statistical allowance. This fact further supports not performing setpoint verification for these relays at power. In addition, the current Surry Technical Specification Definition of CHANNEL FUNCTIONAL TEST does not require setpoint verification:

> "Injection of a simulated signal into an analog channel as close to the sensor as practicable or makeup of the logic combinations in a logic channel to verify that it is operable, including alarm and/or trip initiating action."

NRC Question 3

Provide the basis for the proposed twenty-four hours provided in Action Statement 3.B for the Intermediate Range Flux Instruments.

Dominion Response

The proposed change is acceptable for the following reasons: (a) one intermediate range and the power range channels remain fully OPERABLE and capable of performing the required trip function, (b) requiring a power change either up or down places the plant in a condition where neutron flux protection is operable and ensures additional neutron flux channels are available to monitor reactor power, and (c) although the intermediate range provides a trip function, it is not credited in our safety analyses.

Without the limitation of 24 hours, the reactor could be operated for an indefinite period below P-10 and above P-6 with an inoperable intermediate range. The proposed TS change, which requires reactor power to be reduced below P-6 or increased above the P-10 setpoint within 24 hours with an inoperable channel, is more conservative.

NRC Question 4

Does WCAP-14333 support increasing the allowed outage time (AOT) from 6 to 72 hours for Functional Unit 4, Loss of Power, in Table 3.7-2?

Dominion Response

Although the AOT for this functional unit was permitted to be increased from one hour to six hours in the original Technical Specification Optimization Program (TOPS), this functional unit was not explicitly modeled in the risk analysis performed for WCAP-14333 to increase the AOT to 72 hours. After further review, several additional functional units or their logic configurations were identified as not adequately addressed by the Westinghouse risk analysis performed for the increased AOTs and decreased surveillance frequencies. A plant specific probabilistic risk analysis has been performed to support these changes and is addressed in Attachment 2 to this letter.

Attachment 2
Discussion of Revised 4.1 Basis for
Proposed Technical Specifications and Bases Change - RPS and ESFAS Analog Instrumentation Surveillance Frequency Change

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

Discussion of Change

Introduction

By letter dated June 16, 2000 (Serial No. 00-303), Virginia Electric and Power Company (Dominion) proposed changes to the Surry Technical Specifications. Specifically, changes were requested to the allowed outage time, bypass time, and surveillance interval for the instrumentation for the Reactor Trip (RTS) and Engineered Safeguards Features Actuation Systems (ESFAS) consistent with WCAPs - 10271 and 14333.

The bases for the changes were generic risk evaluations performed by Westinghouse Electric Company for the RTS and ESFAS analog instrumentation as documented in WCAP-10271 Supplements 1 and 2, WCAP-14333, as well as the associated NRC SERs. As part of the proposed changes, Dominion erroneously applied the generic Westinghouse risk evaluation to several instruments that were either not included in the generic evaluation performed or the channel logic was different than modeled by Westinghouse. In order to support the relaxation of the allowed outage times, bypass times and surveillance intervals for these channels (hereafter referred to as *the WCAP changes*) a plant-specific risk assessment was performed.

This package supplements our June 16, 2000 submittal and provides the results of our plant-specific risk assessment for the functional units not addressed in the WCAPs identified above. In addition to the plant-specific risk assessment results, we have included additional discussion of our plant-specific probabilistic risk assessment in the Technical Specifications Basis Section.

Current Licensing Basis

The current surveillance interval for Reactor Protection and Engineered Safety Systems instrumentation, including analog channels, actuation logic and actuation relays is monthly. The allowed outage time for the analog instrument channels is six hours, consistent with WCAP-10271. The minimum testing frequency for those instrument channels is based on an average unsafe failure rate of 2.5 E-6 failure/hr per channel. This failure rate is based on operating experience at conventional and nuclear units through the 1960s.

For the specified one-month test interval, the average unprotected time is 360 hours in case of a failure occurring between test intervals. Thus, the probability of failure of one channel between test intervals is 360 x 2.5 E-6 or 0.9 E-3. Since two channels must fail in order to negate the safety function, the probability of simultaneous failure of two-out-of-three channels is $3(.9 \text{ E-}3)^2 = 2.4 \text{ E-}6$. This represents the fraction of time in which each three-channel system would have one operable and two inoperable channels and equals 2.4 E-6 x 8760 hours per year, or (approximately) 1 minute/year.

Current Design Basis

The reactor protection system provides the means for controlling the reactor in response to various measured primary and secondary variables associated with power, temperature, pressure, level, flow, and the availability of electric power. If the combination of monitored variables indicates an approach to unsafe conditions, the reactor protection system will initiate the appropriate protective action, e.g., load runback, prevention of rod withdrawal, or reactor trip (opening the reactor trip breakers).

The reactor protection system and the engineered safeguards are designed in accordance with IEEE-279, "Nuclear Power Plant Protection Systems," August 1968. The reactor protection system is designed so that the most probable modes of failure in each channel result in a reactor trip signal. The protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not defeat the channel function, cause a spurious trip, or violate reactor protection criteria.

Reactor Protection System channels are designed with sufficient redundancy for individual channel calibration and testing to be performed during power operation without degrading reactor protection. Exceptions are the backup channels such as reactor coolant pump breakers. Removal of one trip channel is accomplished by placing that channel in a trip mode. For example, a two-out-of-three channel becomes a one-out-of-two channel. Testing will not cause a trip unless a trip condition exists concurrently in another channel. During such operation the active parts of the system continue to meet the single-failure criterion, since the channel under test is either tripped or makes use of superimposed test signals that do not negate the process signal. “One-out-of-two” systems are permitted to violate the single-failure criterion during channel bypass provided that acceptable reliability of operation can be otherwise demonstrated and the bypass time interval is short.

Discussion of Change

A plant-specific risk assessment was completed to justify the WCAP changes for the additional functions addressed by this package. The probabilistic risk evaluation assessed the change in core damage frequency (CDF) and the incremental change in core damage probability as a result of the WCAP changes for the additional functions. The following provides a discussion of the risk assessment performed for the functional units not included in the generic WCAP risk assessment.

The CDF sensitivity for the functional units in question was developed in the same manner as the original WCAP-10271 and WCAP-14333 analyses. The EDG start-failure and Non-Essential Service Water isolation function impacts were estimated by fault tree modeling. The Recirculation Mode Transfer (RMT), AFW pump start and containment pressure functions are similar to that of some of the other WCAP channels and were estimated by comparison to similar functions. The RCP breaker position trip is unique and was estimated by combining representative failure probabilities for each of the instrument channel components. Once the channel failure impacts were quantified, these numbers were converted to a CDF impact by looking at the associated CDF sensitivity from the PRA model for the same function or a higher level function.

Reactor Trip on Reactor Coolant Pump Breaker Position (Functional Unit 16, Table 3.7-1) -The reactor trip function initiated on RCP breaker position is not included in the PRA model. However, its unavailability was estimated with and without operator action, both above and below Permissive P-8. Both random and common cause failures were evaluated. At worst, the total signal unavailability is increased by approximately a factor of two by the proposed TS limits. However, the signal unavailability remains very small in every case. When these unavailabilities are used to estimate the risk sensitivity, their net impact is negligible. This latter point is made by noting that the individual logic trains of reactor protection are individually NOT risk-significant. Individual components of the reactor protection system are of proportionally much lower impact.

Containment HI and HI-HI Pressure (Functional Unit 1.b, Safety Injection - containment pressure 3 of 4 logic, Functional Unit 2.b Containment Spray - containment pressure 3 of 4 logic, Table 3.7-2; and Functional Units 1.b.1) and 1.c.1) Containment Isolation – Phases 2 and 3 -containment pressure 3 of 4 logic, Table 3.7-3) - The unavailability of the containment HI and HI-HI functions increases less than a factor of three. However, since the logic trains for these containment pressure functions are extremely low risk, the corresponding CDF impact remains below the limits of roundoff error. The proposed change for these functions has a negligible CDF impact.

Steam-driven Auxiliary Feedwater Pump Start on Reactor Coolant Pump Bus Undervoltage (Functional Unit 3, Table 3.7-2) - The proposed change increases this AFW pump start failure by approximately 1.5%. Converted to a CDF sensitivity, the CDF impact remains below the limits of roundoff error. The proposed change for this function has a negligible impact.

Emergency Diesel Generator Start on Bus Undervoltage/Degraded Voltage (Functional Unit 4, Table 3.7-2) - The EDG start is modeled in the PRA and its CDF impact may be quantified more accurately. Both the undervoltage and the degraded voltage (UV/DV) contributions to the EDG start were evaluated. The net impact of the proposed TS change is an increase in the EDG start-failure probability of approximately 0.5 percent. This failure mode is only marginally risk significant in a zero-maintenance configuration. The increase per EDG in start-failure probability yields a CDF increase of approximately 0.01%.

Nonessential Service Water Isolation on Low Canal Level (Functional Unit 5, Table 3.7-2) - The individual channels for this function see an unavailability increase of approximately 80% due to the proposed WCAP-14333 changes. However, the CDF sensitivity for this function are extremely low, so that the CDF impact remains below the limits of roundoff error. The proposed change has a negligible impact for this function.

Recirculation Mode Transfer (Functional Unit 7, Table 3.7-2) - The RMT function occurs when the Refueling Water Storage Tank level drops to its established setpoint. Its failure probability is estimated to increase by approximately 9.0E-5 as a result of the proposed changes. However, the RMT function has a negligible risk impact in the zero-maintenance configuration. This minor increase in its unavailability also results in a negligible CDF impact.

The additional RPS/ESFAS functions addressed in this package are minor contributors at most to the overall core damage frequency. Their proposed WCAP-14333 changes have only a negligible impact on CDF. These sensitivities are easily bounded by the generic and plant-specific analyses previously reviewed and approved by the NRC for similar functions.

The numbers cited in this review are summarized in the attached table. These numbers account for both the decreased Surveillance Test Interval (STI), and increased Allowed Outage Time (AOT) and bypass time changes.

RG 1.177 outlines specific principles for the implementation of risk-informed Technical Specifications changes. These principles, and a discussion of the Dominion program for meeting their requirements, are as follows.

1. *The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.* The only proposed changes are to the Allowed Outage Times, Bypass Times and Surveillance Test Intervals for the channels addressed in this package. Their basis is explicitly addressed herein.

2. *The proposed change is consistent with the defense-in-depth philosophy. Defense-in-depth is fully maintained.* The redundancy and diversity of the Reactor Protection System is not affected. All components and test requirements remain in place. The only proposed changes are in Allowed Outage Times, Bypass Times and Surveillance Test Intervals.

3. *The proposed changes maintain sufficient safety margins.* The safety margins remain unaffected. The proposed changes are specifically shown to be negligible in every case. No margin of safety is approached for CDF or any other parameter.

4. *When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.* The proposed changes are specifically shown to be on the order of a hundredth of a percent of CDF. NUREG-0800 identifies an incremental conditional core damage probability (ICCDP) of 5E-7 as a threshold for identifying AOT changes as small. The changes proposed in this package are several orders of magnitude less than this limit.

5. *The impact of the proposed change should be monitored using performance measurement strategies.* The existing 10 CFR 50.65 (Maintenance Rule) program monitors the availability and reliability of risk-significant plant components including those in the RPS and ESFAS. Performance is held to stringent criteria and corrective measures are implemented when any component fails to meet its criterion.

Specific Changes

Current Technical Specifications

In addition to the Bases changes proposed in our June 16, 2000 submittal, the following is being proposed to be added to the Bases Section of Section 4.1 to address the plant specific probabilistic risk assessment.

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.

Please substitute the revised bases pages included in Attachment 3 of this submittal to complete your review.

Safety Significance

In WCAP-14333, the WOG evaluated the impact of the additional relaxation of allowed outage times and completion times, and action statements on core damage frequency. The associated change in core damage frequency is an increase of 3.1 percent for those plants with two out of three logic schemes that have not implemented the proposed changes evaluated in WCAP-10271 and its supplements. The NRC Staff considered this resultant core damage frequency (CDF) increase to be small compared to the range of uncertainty in the core damage frequency analyses, and therefore found it acceptable.

The NRC performed an independent, generic evaluation of the impact on CDF and large early release frequency (LERF). The results of the staff's review indicate that the increase in core damage frequency is small (approximately 3.2%) and the large early release fraction would increase by only 4 percent for 2 out of 3 logic schemes that have not implemented the proposed changes evaluated in WCAPs.

Dominion's original evaluation used the current Surry PRA model to establish an overall change in the CDF of approximately one percent in a plant-specific analysis. This result is consistent with the original WCAP–10271 and WCAP-14333 analyses. The combined impact of the supplemental changes addressed by this package is only approximately a hundredth of a percent, two orders of magnitude less than the original June 16, 2000 submittal result. The impact on the incremental core damage probability is several orders of magnitude smaller. These numbers for the individual channels are also consistent with the channel-specific analyses of the WCAPs. Thus, the generic WCAP analyses remains bounding and Dominion's plant-specific analysis remains unaffected by the additional functions. The overall impact on Surry CDF due to implementation of WCAP-10271 and WCAP-14333 relaxations in allowed outage time, bypass time, and surveillance interval for RTS and ESFAS instrumentation is minor.

Table 1 – Risk Evaluation Results

FUNCTION	Nominal Unavailability	Potential Unavailability Change *	Percent Change	Risk Reduction Worth	Risk Achievement Worth	CDF Impact **	Comment
Reactor Trip on RCP Breaker Position	8.94E-05	9.86E-05	110%	--	--	--	Above P-8
	3.61E-04	8.09E-04	224%	--	--	--	Below P-8
Reactor Trip Logic Trains	1.18E-03	--	--	1	1.00	0.00%	Below roundoff error
HI and HI-HI Containment Pressure	9.44E-04	1.58E-03	167%	1.000	1.00	0.00%	Below roundoff error
AFW Pump Start on RCP Bus Undervoltage	2.51E-02	3.68E-04	1.5%	1.002	1.09	0.00%	Below roundoff error
EDG Auto-start	3.98E-02	0.02E-02	0.5%	1.012	2.52	0.01%	Accounts for unit specific and swing EDGs
Nonessential Service Water Isolation on Low Canal Level	5.92E-04	4.58E-04	77%	1.000	1.21	0.00%	Below roundoff error
Recirculation Mode Transfer	7.21E-05	8.99E-05	125%	1.000	2.65	0.00%	Below roundoff error

* These numbers are based upon the assumption that the full AOT will be used on a regular basis every year. In fact, these functions are rarely removed from service during power operation.

** The baseline risk for Surry is presently quantified at approximately 3.0E-5/yr.

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.