UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 29, 2011

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING RELOCATION OF SURVEILLANCE FREQUENCIES TO LICENSEE-CONTROLLED PROGRAM USING RISK-INFORMED JUSTIFICATION (TSTF-425) (TAC NOS. ME3687 AND ME3688)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 273 to Renewed Facility Operating License No. DPR-32 and Amendment No. 272 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments revise the Technical Specifications (TSs) in response to your application dated March 30, 2010, as supplemented by letters dated August 23, 2010, and March 4, 2011.

These amendments revise the TSs by relocating specific surveillance frequencies to a licensee-controlled document using a risk-informed justification.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

For

Karen Cotton, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 273 to DPR-32
2. Amendment No. 272 to DPR-37
3. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 273
Renewed License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated March 30, 2010, as supplemented by letters dated August 23, 2010, and March 4, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 273 , are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-32
and the Technical Specifications

Date of Issuance: April 29, 2011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 272
Renewed License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated March 30, 2010, as supplemented by letters dated August 23, 2010, and March 4, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 272 , are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

G (C

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes License No. DPR-37
and the Technical Specifications

Date of Issuance : April 29, 2011

ATTACHMENT

TO LICENSE AMENDMENT NO. 273

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

TO LICENSE AMENDMENT NO. 272

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
License No. DPR-32, page 3	License No. DPR-32, page 3
License No. DPR-37, page 3	License No. DPR-37, page 3
TSs	TSs
TS 4.1-1	TS 4.1-1
TS 4.1-1a	TS 4.1-1a
TS 4.1-1b	TS 4.1-1b
TS 4.1-3	TS 4.1-3
TS 4.1-4	TS 4.1-4
TS 4.1-5	TS 4.1-5
TS 4.1-5a	TS 4.1-5a
TS 4.1-6	TS 4.1-6
TS 4.1-7	TS 4.1-7
TS 4.1-8	TS 4.1-8
TS 4.1-8a	TS 4.1-8a
TS 4.1-8b	TS 4.1-8b
TS 4.1-8c	TS 4.1-8c
TS 4.1-9	TS 4.1-9
TS 4.1-9a	TS 4.1-9a
TS 4.1-9b	TS 4.1-9b
TS 4.1-9c	TS 4.1-9c
TS 4.1-9d	TS 4.1-9d
TS 4.1-10	TS 4.1-10

Remove Pages	Insert Pages
TS 4.5.2	TS 4.5.2
TS 4.5-3	TS 4.5-3
TS 4.5-4	TS 4.5-4
TS 4.6-1	TS 4.6-1
TS 4.6-2	TS 4.6-2
TS 4.6-3	TS 4.6-3
TS 4.6-4	TS 4.6-4
TS 4.6-5	TS 4.6-5
TS 4.8-1	TS 4.8-1
TS 4.8-2	TS 4.8-2
TS 4.8-3	TS 4.8-3
TS 4.8-4	TS 4.8-4
TS 4.9-1	TS 4.9-1
TS 4.10-1	TS 4.10-1
TS 4.10-2	TS 4.10-2
TS 4.10-3	TS 4.10-3
TS 4.11-1	TS 4.11-1
TS 4.11-2	TS 4.11-2
TS 4.11-3	TS 4.11-3
TS 4.11-4	TS 4.11-4
TS 4.12-1	TS 4.12-1
TS 4.12-2	TS 4.12-2
TS 4.12-3	TS 4.12-3
TS 4.12-4	TS 4.12-4
TS 4.12-6	TS 4.12-6
TS 4.13-1	TS 4.13-1
TS 4.13-2	TS 4.13-2
TS 4.13-2a	TS 4.13-2a
TS 4.16-2	TS 4.16-2
TS 4.18-1	TS 4.18-1
TS 6.4-15	TS 6.4-15

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

   The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 273 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. Reports

   The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

   The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 71

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

   I. Fire Protection

   The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 272 , are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Deleted by Amendment 54

F. Deleted by Amendment 59 and Amendment 65

G. Deleted by Amendment 227

H. Deleted by Amendment 227

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

A. Calibration, testing, and checking of instrumentation channels and interlocks shall be performed as detailed in Tables 4.1-1, 4.1-1A, and 4.1-2 and at the frequencies specified in the Surveillance Frequency Control Program, unless otherwise noted in the Tables.

B. Equipment tests shall be performed as detailed in Table 4.1-2A and at the frequencies specified in the Surveillance Frequency Control Program, unless otherwise noted in the Tables and as detailed below.

1. In addition to the requirements of the Inservice Testing Program, each Pressurizer PORV and block valve shall be demonstrated OPERABLE at the frequencies specified in the Surveillance Frequency Control Program by:

   a. Performing a complete cycle of each PORV with the reactor coolant average temperature >350°F.

   b. Performing a complete cycle of the solenoid air control valve and check valves on the air accumulators in the PORV control system.

   c. Operating each block valve through one complete cycle of travel. This surveillance is not required if the block valve is closed in accordance with 3.1.6.a, b, or c.

   d. Verifying that the pressure in the PORV backup air supply is greater than the surveillance limit.

   e. Performing functional testing and calibration of the PORV backup air supply instrumentation and alarm setpoints.

2. The pressurizer water volume shall be determined to be within its limit as defined in Specification 2.3.A.3.a at least once per 12 hours whenever the reactor is not subcritical by at least 1% $\Delta k/k$.

3. Each Reactor Vessel Head vent path remote operating isolation valve not required to be closed by Specification 3.1.A.7a or 3.1.A.7b shall be demonstrated OPERABLE at each COLD SHUTDOWN but not more often than once per 92 days by operating the valve through one complete cycle of full travel from the control room.

4. Each Reactor Vessel Head vent path shall be demonstrated OPERABLE following each refueling by:

   a. Verifying the manual isolation valves in each vent path are locked in the open position.

   b. Cycling each remote operating isolation valve through at least one complete cycle of full travel from the control room.

   c. Verifying flow through the reactor vessel head vent system vent paths.

C. Sampling tests shall be conducted as detailed in Table 4.1-2B and at the frequencies specified in the Surveillance Frequency Control Program, unless otherwise noted in the Table.

D. Whenever containment integrity is not required, only the asterisked items in Table 4.1-1 and 4.1-2A and 4.1-2B are applicable.

E. Flushing of wetted sensitized statinless steel pipe sections as identified in the Basis Section shall be conducted only if the RWST Water Chemistry exceeds 0.15 PPM chlorides and/or fluorides (CL^- and or F^-). Flushing of sensitized stainless steel pipe sections shall be conducted as detailed in TS Table 4.1-3A and 4.1-3B.

F. Containment Ventilation Purge System isolation valves:

1. The outside Containment Ventilation Purge System isolation valves and the isolation valve in the containment vacuum ejector suction line outside containment shall be determined locked, sealed, or otherwise secured in the closed position at the frequency specified in the Surveillance Frequency Control Program.

2. The inside Containment Ventilation Purge System isolation valves and the isolation valve in the containment vacuum ejector suction line inside containment shall be verified locked, sealed, or otherwise secured in the closed position each COLD SHUTDOWN, but not required to be verified more than once per 92 days.

G. Verify that each containment penetration not capable of being closed by OPERABLE automatic isolation valves and required to be closed during accident conditions is closed by manual valves, blind flanges, or deactivated automatic valves secured* in the closed position at the frequency specified in the Surveillance Frequency Control Program. Valves, blind flanges, and deactivated automatic or manual valves located inside containment which are locked, sealed, or otherwise secured in the closed position shall be verified closed during each COLD SHUTDOWN, but not required to be verified more than once per 92 days.

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* Non-automatic or deactivated automatic valves may be opened on an intermittent basis under administrative control. The valves identified in TS 3.8.A.2 and TS 3.8.A.3 are excluded from this provision.

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 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.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

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 test frequencies are controlled under the Surveillance Frequency Control Program.

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 at the frequency specified in the Surveillance Frequency Control Program. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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 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

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	SFCP	SFCP (1,5) SFCP (3,5) SFCP(4)	SFCP (2)	1) Against a heat balance standard, above 15% RATED POWER 2) Signal at ΔT; bistable action (permissive, rod stop, trip) 3) Upper and lower chambers for symmetric offset by means of the movable incore detector system 4) Neutron detectors may be excluded from CHANNEL CALIBRATION 5) The provisions of Specification 4.0.4 are not applicable
2.	Nuclear Intermediate Range (below P-10 setpoint)	*SFCP	SFCP (2,3)	P(1)	1) Log level; bistable action (permissive, rod stop, trip) 2) Neutron detectors may be excluded from CHANNEL CALIBRATION 3) The provisions of Specification 4.0.4 are not applicable
3.	Nuclear Source Range (below P-6 setpoint)	*SFCP	SFCP (2,3)	P(1)	1) Bistable action (alarm, trip) 2) Neutron detectors may be excluded from CHANNEL CALIBRATION 3) The provisions of Specification 4.0.4 are not applicable
4.	Reactor Coolant Temperature	*SFCP	SFCP	SFCP (1) SFCP (2)	1) Overtemperature ΔT 2) Overpower ΔT
5.	Reactor Coolant Flow	SFCP	SFCP	SFCP	
6.	Pressurizer Water Level	SFCP	SFCP	SFCP	
7.	Pressurizer Pressure (High & Low)	SFCP	SFCP	SFCP	
8.	4 KV Voltage and Frequency	N.A.	SFCP	SFCP (1)	1) Setpoint verification not required
9.	Analog Rod Position	*SFCP (1,2) (3)	SFCP	N.A.	1) With step counters 2) Each six inches of rod motion when data logger is out of service 3) N.A. when reactor is in HOT, INTERMEDIATE OR COLD SHUTDOWN

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	SFCP (1,2) SFCP (3)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position 3) For the control banks, the benchboard indicators shall be checked against the output of the bank overlap unit.
11.	Steam Generator Level	SFCP	SFCP	SFCP	
12.	Deleted				
13.	Deleted				
14.	Deleted				
15.	Recirculation Mode Transfer				
	a. Refueling Water Storage Tank Level-Low-Low	SFCP	SFCP	SFCP	
	b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	SFCP	
16.	Recirculation Spray Pump Start				
	a. RWST Level-Low	SFCP	SFCP	SFCP	
17.	Reactor Containment Pressure-CLS	*SFCP	SFCP	SFCP (1)	1) Isolation valve signal and spray signal
18.	Deleted				
19.	Deleted				
20.	Deleted				
21.	Deleted				
22.	Steam Line Pressure	SFCP	SFCP	SFCP	

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
23.	Turbine First Stage Pressure	SFCP	SFCP	SFCP	
24.	Deleted				
25.	Deleted				
26.	Logic Channel Testing	N.A.	N.A.	SFCP (1)(2)	1) Reactor protection, safety injection and the consequence limiting safeguards system logic are tested per this line item. 2) The master and slave relays are not included in the periodic logic channel test of the safety injection system.
27.	Deleted				
28.	Turbine Trip				Setpoint verification is not applicable
	a. Stop valve closure	N.A.	N.A.	P	
	b. Low fluid oil pressure	N.A.	N.A.	P	
29.	Deleted				
30.	Reactor Trip Breaker	N.A.	N.A.	SFCP	The test shall independently verify operability of the undervoltage and shunt trip attachments
31.	Deleted				

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
32. Auxiliary Feedwater				
a. Steam Generator Water Level Low-Low	SFCP	SFCP	SFCP (1)	1) The auto start of the turbine driven pump is not included in the periodic test, but is tested within 31 days prior to each startup.
b. RCP Undervoltage	SFCP	SFCP	SFCP (1)(2)	1) The actuation logic and relays are tested within 31 days prior to each startup. 2) Setpoint verification not required.
c. S.I.	(All Safety Injection surveillance requirements)			
d. Station Blackout	N.A.	SFCP	N.A.	
e. Main Feedwater Pump Trip	N.A.	N.A.	SFCP	
33. Loss of Power				
a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	SFCP	SFCP (1)	1) Setpoint verification not required.
b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	SFCP	SFCP (1)	1) Setpoint verification not required.
34. Deleted				
35. Manual Reactor Trip	N.A.	N.A.	SFCP	The test shall independently verify the operability of the undervoltage and shunt trip attachments for the manual reactor trip function. The test shall also verify the operability of the bypass breaker trip circuit.
36. Reactor Trip Bypass Breaker	N.A.	N.A.	SFCP (1), SFCP (2)	1) Remote manual undervoltage trip immediately after placing the bypass breaker into service, but prior to commencing reactor trip system testing or required maintenance. 2) Automatic undervoltage trip.
37. Safety Injection Input to RPS	N.A.	N.A.	SFCP	
38. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	SFCP	

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
39.	Steam/Feedwater Flow and Low S/G Water Level	SFCP	SFCP	SFCP (1)	1) The provisions of Specification 4.0.4 are not applicable
40.	Intake Canal Low (See Note 1)	SFCP	SFCP	SFCP (1), SFCP (2)	1) Logic Test 2) Channel Electronics Test
41.	Turbine Trip and Feedwater Isolation				
	a. Steam generator water level high	SFCP	SFCP	SFCP	
	b. Automatic actuation logic and actuation relay	N.A.	SFCP	SFCP (1)	1) Automatic actuation logic only, actuation relays tested each refueling
42.	Reactor Trip System Interlocks				
	a. Intermediate range neutron flux, P-6	N.A.	SFCP (1)	SFCP (2)	1) Neutron detectors may be excluded from the calibration
	b. Low reactor trips block, P-7	N.A.	SFCP (1)	SFCP (2)	2) The provisions of Specification 4.0.4 are not applicable.
	c. Power range neutron flux, P-8	N.A.	SFCP (1)	SFCP (2)	
	d. Power range neutron flux, P-10	N.A.	SFCP (1)	SFCP (2)	
	e. Turbine impulse pressure	N.A.	SFCP	SFCP	

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
43.	Engineered Safeguards Actuation Interlocks				
	a. Reactor trip, P-4	N.A.	N.A.	SFCP	
	b. Pressurizer pressure, P-11	N.A.	SFCP	SFCP	
	c. Low, low T_{avg}, P-12	N.A.	SFCP	SFCP	

P - Prior to each startup if not done within the frequency specified in the Surveillance Frequency Control Program

SFCP - Surveillance frequencies are specified in the Surveillance Frequency Control Program.

Note 1:

Check — Consists of verifying for an indicated intake canal level greater than 23'-5.85" that all four low level sensor channel alarms are not in an alarm state.

Calibration — Consists of uncovering the level sensor and measuring the time response and voltage signals for the immersed and dry conditions. It also verifies the proper action of instrument channel from sensor to electronics to channel output relays and annunciator. Only the two available sensors on the shutdown unit would be tested.

Tests
1) The logic test verifies the three out of four logic development for each train by using the channel test switches for that train.
2) Channel electronics test verifies that electronics module responds properly to a superimposed differential millivolt signal which is equivalent to the sensor detecting a "dry" condition.

TABLE 4.1-1A
EXPLOSIVE GAS MONITORING INSTRUMENTATION REQUIREMENTS

CHANNEL DESCRIPTION	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Waste Gas Holdup System Explosive Gas Monitoring System			
Oxygen Monitor	SFCP	SFCP (1)	SFCP

SFCP - Surveillance frequencies are specified in the Surveillance Frequency Control Program.

(1) The channel calibration shall include the use of standard gas samples containing a nominal:

1. one volume percent oxygen, balance nitrogen, and

2. four volume percent oxygen, balance nitrogen

TABLE 4.1-2

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK (1)	CHANNEL CALIBRATION
1.	Auxiliary Feedwater Flow	SFCP	SFCP
2.	Inadequate Core Cooling	SFCP	SFCP
3.	Containment Pressure (Wide Range)	SFCP	SFCP
4.	Containment Pressure	SFCP	SFCP
5.	Containment Sump Water Level (Wide Range)	SFCP	SFCP
6.	Containment Area Radiation (High Range)	SFCP	SFCP
7.	Power Range Neutron Flux	SFCP	SFCP (2)
8.	Source Range Neutron Flux	SFCP	SFCP (2)
9.	Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	SFCP	SFCP
10.	RCS Cold Leg Temperature (Wide Range)	SFCP	SFCP
11.	RCS Pressure (Wide Range)	SFCP	SFCP
12.	Penetration Flow Path Containment Isolation Valve Position	SFCP	SFCP (3)
13.	Pressurizer Level	SFCP	SFCP
14.	Steam Generator (SG) Water Level (Wide Range)	SFCP	SFCP
15.	SG Water Level (Narrow Range)	SFCP	SFCP
16.	SG Pressure	SFCP	SFCP
17.	Emergency Condensate Storage Tank Level	SFCP	SFCP
18.	High Head Safety Injection Flow to Cold Leg	SFCP	SFCP

SFCP - Surveillance frequencies are specified in the Surveillance Frequency Control Program.

(1) Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.

(2) Neutron detectors are excluded from CHANNEL CALIBRATION.

(3) Rather than CHANNEL CALIBRATION, this surveillance shall be an operational test, consisting of verification of operability of all devices in the channel.

TABLE 4.1-2A

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
1.	Control Rod Assemblies	Rod drop times of all full length rods at hot conditions	Prior to reactor criticality: a. For all rods following each removal of the reactor vessel head b. For specially affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods c. SFCP	7
2.	Control Rod Assemblies	Partial movement of all rods	SFCP	7
3.	Refueling Water Chemical Addition Tank	Functional	SFCP	6
4.	Pressurizer Safety Valves	Setpoint	Per the Inservice Testing Program	4
5.	Main Steam Safety Valves	Setpoint	Per the Inservice Testing Program	10
6.	Containment Isolation Trip	* Functional	SFCP	5
7.	Refueling System Interlocks	* Functional	Prior to refueling	9.12
8.	Service Water System	* Functional	SFCP	9.9
9.	Deleted			
10.	Deleted			
11.	Diesel Fuel Supply	* Fuel Inventory	SFCP	8.5
12.	Deleted			
13.	Main Steam Line Trip Valves	Functional (Full Closure)	Before each startup (TS 4.7) The provisions of Specification 4.0.4. are not applicable	10

TABLE 4.1-2A (CONTINUED)
MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
14a.	Service Water System Valves in Line Supplying Recirculation Spray Heat Exchangers	Functional	SFCP	9.9
b.	Service Water System Valves Isolating Flow to Non-essential loads on Intake Canal Low Level Isolation	Functional	SFCP	9.9
15.	MCR/ESGR Envelope Isolation Actuation Instrumentation - Manual	Functional	SFCP	9.13
16.	Reactor Vessel Overpressure Mitigating System (except backup air supply)	Functional & Setpoint	Prior to decreasing RCS temperature below 350°F and monthly while the RCS is < 350°F and the Reactor Vessel Head is bolted	4.3
		CHANNEL CALIBRATION	SFCP	
17.	Reactor Vessel Overpressure Mitigating System Backup Air Supply	Setpoint	SFCP	4.3
18.	Power-Operated Relief Valve Control System	Functional, excluding valve actuation	SFCP	4.3
		CHANNEL CALIBRATION	SFCP	

TABLE 4.1-2A(CONTINUED)
MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST		FREQUENCY	UFSAR SECTION REFERENCE
19.	Primary Coolant System	Functional	1.	Periodic leakage testing(a)(b) on each valve listed in Specification 3.1.C.5.a shall be accomplished prior to entering POWER OPERATION after every time the plant is placed in COLD SHUTDOWN for refueling, after each time the plant is placed in COLD SHUTDOWN for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
20.	Containment Purge MOV Leakage	Functional		Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval(c)	
21.	Deleted				
22.	RCS Flow	Flow ≥ 273,000 gpm and ≥ the limit as specified in the CORE OPERATING LIMITS REPORT		SFCP (d)	14
23.	Deleted				

SFCP - Surveillance frequencies are specified in the Surveillance Frequency Control Program.

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

(d) Not required to be performed until 7 days after ≥ 90% RATED POWER.

* See Specification 4.1.D.

TABLE 4.1-2B
MINIMUM FREQUENCIES FOR SAMPLING TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
1.	Reactor Coolant Liquid Samples	Radio-Chemical Analysis (1)	SFCP (5)	
		Gross Activity (2)	SFCP (5)	9.1
		Tritium Activity	SFCP (5)	9.1
		* Chemistry (CL, F & O_2)	SFCP (9)	4
		* Boron Concentration	SFCP	9.1
		$\bar{E}$ Determination	SFCP (3)	
		DOSE EQUIVALENT I-131	SFCP (5)	
		Radio-iodine Analysis (including I-131, I-133 & I-135)	Once/4 hours (6) and (7) below	
2.	Refueling Water Storage	Chemistry (Cl & F)	SFCP	6
3.	Boric Acid Tanks	* Boron Concentration	SFCP	9.1
4.	Chemical Additive Tank	NaOH Concentration	SFCP	6
5.	Spent Fuel Pit	* Boron Concentration	SFCP	9.5
6.	Secondary Coolant	DOSE EQUIVALENT I-131	SFCP	
7.	Stack Gas Iodine and Particulate Samples	* I-131 and particulate radioactive releases	SFCP	

* See Specification 4.1.D

SFCP - Surveillance frequencies are specified in the Surveillance Frequency Control Program.

(1) A radiochemical analysis will be made to evaluate the following corrosion products: Cr-51, Fe-59, Mn-54, Co-58, and Co-60.

(2) A gross beta-gamma degassed activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc.

2. By verifying that each motor-operated valve in the recirculation spray flow paths performs satisfactorily when tested in accordance with the Inservice Testing Program.

3. By verifying each spray nozzle is unobstructed following maintenance which could cause nozzle blockage.

C. In addition to the requirements of the Inservice Testing Program, each weight-loaded check valve in the containment spray and outside containment recirculation spray subsystems shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by cycling the valve one complete cycle of full travel and verifying that each valve opens when the discharge line of the pump is pressurized with air and seats when a vacuum is applied.

D. Verify, by visual inspection at the frequency specified in the Surveillance Frequency Control Program, that the recirculation spray containment sump components are not restricted by debris and show no evidence of structural distress or abnormal corrosion.

Basis

The flow testing of each containment spray pump is performed by opening the normally closed valve in the containment spray pump recirculation line returning water to the refueling water storage tank. The containment spray pump is operated and a quantity of water recirculated to the refueling water storage tank. The discharge to the tank is divided into two fractions; one for the major portion of the recirculation flow and the other to pass a small quantity of water through test nozzles which are identical with those used in the containment spray headers.

The purpose of the recirculation through the test nozzles is to assure that there are no particulate material in the refueling water storage tank small enough to pass through pump suction strainers and large enough to clog spray nozzles.

Due to the physical arrangement of the recirculation spray pumps inside the containment, it is impractical to flow-test them other than during a unit outage. Flow testing of these pumps requires the physical modification of the pump discharge piping and the erection of a temporary dike to contain recirculated water. The length of time required to setup for the test, perform the test, and then reconfigure the system for normal operation is prohibitive to performing the flow-test on even the cold shutdown frequency. Therefore, the flow-test of the inside containment recirculation spray pumps will be performed in accordance with the Inservice Testing Program during a unit outage.

The inside containment recirculation spray pumps are capable of being operated dry for approximately 60 seconds without significantly overheating and/or degrading the pump bearings. During this dry pump check, it can be determined that the pump shafts are turning by rotation sensors which indicate in the Main Control Room. In addition, motor current will be compared with an established reference value to ascertain that no degradation of pump operation has occurred.

The recirculation spray pumps outside the containment have the capability of being dry-run and flow tested. The test of an outside recirculation spray pump is performed by closing the containment sump suction line valve and the isolation valve between the pump discharge and the containment penetration. This allows the pump casing to be filled with water and the pump to recirculate water through a test line from the pump discharge to the pump casing.

With a system flush conducted to remove particulate matter prior to the installation of spray nozzles and with corrosion resistant nozzles and piping, it is not considered credible that a significant number of nozzles would plug during the life of the unit to reduce the effectiveness of the subsystems. Therefore, an inspection or air or smoke test of the nozzles following maintenance which could cause nozzle blockage is sufficient to indicate that plugging of the nozzles has not occurred.

The spray nozzles in the refueling water storage tank provide means to ensure that there is no particulate matter in the refueling water storage tank and the containment spray subsystems which could plug or cause deterioration of the spray nozzles. The nozzles in the tank are identical to those used on the containment spray headers. The flow test of the containment spray pumps and recirculation to the refueling water storage will indicate any plugging of the nozzles by a reduction of flow through the nozzles.

Periodic inspections of containment sump components ensure that the components are unrestricted and stay in proper operating condition. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

References

FSAR Section 6.3.1, Containment Spray Pumps

FSAR Section 6.3.1, Recirculation Spray Pumps

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTING

Applicability

Applies to periodic testing and surveillance requirements of the Emergency Power System.

Objective

To verify that the Emergency Power System will respond promptly and properly when required.

Specification

The following tests and surveillance shall be performed as stated:

A. Diesel Generators

1. Tests and Frequencies

   a. Manually initiated start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to 2750 Kw. This test will be conducted at the frequency specified in the Surveillance Frequency Control Program on each diesel generator for a duration of 30 minutes. Normal station operation will not be affected by this test.

b. Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment, initiated by a simulated loss of off-site power together with a simulated safety injection signal. Testing will demonstrate load shedding and load sequencing initiated by a simulated loss of off-site power following a simulated engineered safety features signal. Testing will also demonstrate that the loss of voltage and degraded voltage protection is defeated whenever the emergency diesel is the sole source of power to an emergency bus and that this protection is automatically reinstated when the diesel output breaker is opened. This test will be conducted at the frequency specified in the Surveillance Frequency Control Program to assure that the diesel generator will start and accept load in less than or equal to 10 seconds after the engine starting signal.

c. Availability of the fuel oil transfer system shall be verified by operating the system in conjunction with TS 4.6.A.1.a surveillance.

d. Each diesel generator shall be given a thorough inspection at the frequency specified in the Surveillance Frequency Control Program utilizing the manufacturer's recommendations for this class of stand-by service.

2. Acceptance Criteria

The above tests will be considered satisfactory if all applicable equipment operates as designed.

B. Fuel Oil Storage Tanks for Diesel Generators

1. A minimum fuel oil storage of 35,000 gal shall be maintained on-site to assure full power operation of one diesel generator for seven days.

C. Station Batteries

1. Tests and Frequencies

The following Tests shall be performed at the frequencies specified in the Surveillance Frequency Control Program:

a. Measure the specific gravity, electrolytic temperature, cell voltage of the pilot cell in each battery, and the D.C. bus voltage of each battery.

b. Measure the voltage of each battery cell in each battery to the nearest 0.01 volts.

c. Measure the specific gravity of each battery cell, the temperature reading of every fifth cell, the height of electrolyte of each cell, and the amount of water added to any cell.

d. Compare the battery voltage and current after the battery charger has been turned off for approximately 5 min during normal operation.

e. Perform a simulated load test without battery charger on each station battery. The battery voltage and current as a function of time shall be monitored.

f. Check the battery connections for tightness and apply anti-corrosion coating to the interconnections.

2. Acceptance Criteria

a. Each test shall be considered satisfactory if the new data when compared to the old data indicate no signs of abuse or deterioration.

b. The load test in (d) and (e) above shall be considered satisfactory if the batteries perform within acceptable limits as established by the manufacturers discharge characteristic curves.

D. EMERGENCY DIESEL GENERATOR BATTERIES

1. TESTS AND FREQUENCIES

The following Tests shall be performed at the frequencies specified in the Surveillance Frequency Control Program:

a. Measure the specific gravity, electrolytic temperature, cell voltage of the pilot cell in each battery and the D.C. bus voltage of each battery.

b. Measure the voltage of each battery cell in each battery to the nearest 0.01 volts.

c. Measure the specific gravity of each battery cell, the temperature reading of every fifth cell, the height of electrolyte of each cell, and the amount of water added to any cell.

d. Perform a normal load or simulated load test without battery charger on each battery. The battery voltage and current as a function of time shall be monitored.

e. Check the battery connections for tightness and apply anti-corrosion coating to interconnections.

2. ACCEPTANCE CRITERIA

a. Each test shall be considered satisfactory if the new data when compared to the old data indicate no signs of abuse or deterioration.

b. The load test in (d) above shall be considered satisfactory if the batteries perform within acceptable limits as established by the manufacturers discharge characteristic curves.

Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of essential safeguards equipment. They also assure that the emergency diesel generator system controls and the control systems for the safeguards equipment will function automatically in the event of a loss of normal station service power.

The testing frequency specified in the Surveillance Frequency Control Program will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply and starting circuits and controls are continuously monitored and any faults are alarm indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

Station and emergency diesel generator batteries may deteriorate with time, but precipitous failure is extremely unlikely. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. In addition alarms have been provided to indicate low battery voltage and low current from the inverters which would make it extremely unlikely that deterioration would go unnoticed.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the ampere-hour capability of the battery. As a check upon the effectiveness of the equalizing charge, the battery shall be loaded rather heavily and the voltage monitored as a function of time. If a cell has deteriorated or if a connection is loose, the voltage under load will drop excessively indicating the need for replacement or maintenance. FSAR Section 8.5 provides further amplification of the basis.

References

FSAR Section 8.5 Emergency Power System

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to the periodic testing requirements of the Auxiliary Feedwater System.

Objective

To verify the operability of the auxiliary feedwater pumps.

Specification

A. Tests and Frequencies

The following Tests shall be performed at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted below:

1. Verify that the Auxiliary Feedwater System manual, power operated, and automatic valves in each flowpath are in the correct position. This verification includes valves that are not locked, sealed, or otherwise secured in position, valves in the cross-connect from the opposite unit and valves in the steam supply paths to the turbine driven auxiliary feedwater pump.

2. Verify that each motor-operated valve in the auxiliary feedwater flowpaths, including the cross-connect from the opposite unit, performs satisfactorily when tested in accordance with the Inservice Testing Program.

3. Verify that the auxiliary feedwater pumps perform satisfactorily when tested in accordance with the Inservice Testing Program. The provisions of Specification 4.0.4 are not applicable for the turbine driven pump. Note that the developed head test of the turbine driven pump is required to be performed within 24 hours after reaching HOT SHUTDOWN.

4. Whenever the unit's Reactor Coolant System temperature and pressure have been less than 350°F and 450 psig, respectively, for a period greater than 30 days, prior to Reactor Coolant System temperature and pressure exceeding 350°F and 450 psig, respectively, verify proper alignment of the required auxiliary feedwater flowpaths by verifying flow from the 110,000 gallon above ground Emergency Condensate Storage Tank to the steam generators from each of the auxiliary feedwater pumps.

5. During periods of reactor shutdown with the opposite unit's Reactor Coolant System temperature and pressure greater than 350°F and 450 psig, respectively:

   a. Continue to verify that the motor driven auxiliary feedwater pumps perform satisfactorily when tested at the frequency defined in Specification 4.8.A.3.

   b. Verify that each motor-operated valve in the auxiliary feedwater cross-connect flowpath for the opposite unit performs satisfactorily when tested in accordance with the Inservice Testing Program.

6. Verify automatic actuation of:

   a. Each auxiliary feedwater automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

   b. Each auxiliary feedwater pump starts automatically on an actual or simulated actuation signal. Note that this surveillance is required to be performed for the turbine driven pump within 24 hours after reaching HOT SHUTDOWN.

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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 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 in accordance with the Inservice Testing Program 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.

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* 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

4.9 RADIOACTIVE GAS STORAGE MONITORING SYSTEM

Applicability

Applies to the periodic monitoring of radioactive gas storage.

Objective

To ascertain that waste gas is stored in accordance with Specification 3.11.

Specification

A. The concentration of oxygen in the waste gas holdup system shall be determined to be within the limits of Specification 3.11.A by continuously monitoring the waste gases in the waste gas holdup system with the oxygen monitor required to be OPERABLE by Table 3.7-5(a) of Specification 3.7.E.

B. The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.11.B at the frequency specified in the Surveillance Frequency Control Program when the specific activity of the primary reactor coolant is ≤ 2200 μCi/gm dose equivalent Xe-133. Under the conditions which result in a specific activity > 2200 μCi/gm dose equivalent Xe-133, the waste gas decay tanks shall be sampled once per day.

4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared with the predicted value at the frequency specified in the Surveillance Frequency Control Program. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made. The provisions of Specification 4.0.4 are not applicable.

B. During periods of POWER OPERATION at greater than 10% of RATED POWER, the hot channel factors identified in Section 3.12 shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed their limits, an evaluation as to the cause of the anomaly shall be made. The provisions of Specification 4.0.4 are not applicable.

DELETED

Basis

BORON CONCENTRATION

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration necessary to maintain adequate control characteristics must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod assembly groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration, and the slope of the curve relating burnup and reactivity is compared with that predicted. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive control rod assembly in the fully withdrawn position is always maintained.

PEAKING FACTORS

A thermal criterion in the reactor core design specified that "no fuel melting during any anticipated normal operating condition" should occur. To meet the above criterion during a thermal overpower of 118% with additional margin for design uncertainties, a steady state maximum linear power is selected. This then is an upper linear power limit determined by the maximum central temperature of the hot pellet.

The peaking factor is a ratio taken between the maximum allowed linear power density in the reactor to the average value over the whole reactor. It is of course the average value that determines the operating power level. The peaking factor is a constraint which must be met to assure that the peak linear power density does not exceed the maximum allowed value.

During normal reactor operation, measured peaking factors should be significantly lower than design limits. As core burnup progresses, measured designed peaking factors typically decrease. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

4.11 SAFETY INJECTION SYSTEM TESTS

Applicability

Applies to the operational testing of the Safety Injection System.

Objective

To verify that the Safety Injection System will respond promptly and perform its design functions, if required.

Specifications

A. The refueling water storage tank (RWST) shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the RWST solution temperature is within specified limits.
2. Verifying:
    a. The RWST contained borated water volume, and
    b. The RWST boron concentration are within specified limits.

B. Each safety injection accumulator shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program or as specified below by:

1. Verifying:
    a. The contained borated water volume, and
    b. The nitrogen cover-pressure are within specified limits.

2. Verifying:

   a. The boron concentration of the accumulator solution is within specified limits, and

   b. The boron concentration of the accumulator solution within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume.

   Note: Surveillance 4.11.B.2.b is not required when the volume increase makeup source is the RWST.

C. Each Safety Injection Subsystem shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted below by:

1. Verifying, that on recirculation flow, each low head safety injection pump performs satisfactorily when tested in accordance with the Inservice Testing Program.

2. Verifying that each charging pump performs satisfactorily when tested in accordance with the Inservice Testing Program.

3. Verifying that each motor-operated valve in the safety injection flow path performs satisfactorily when tested in accordance with the Inservice Testing Program.

4. Prior to POWER OPERATION by:

   a. Verifying that the following motor operated valves are blocked open by de-energizing AC power to the valves motor operator and tagging the breaker in the off position:

Unit 1	Unit 2
MOV-1890C	MOV-2890C

   b. Verifying that the following motor operated valves are blocked closed by de-energizing AC power to the valves motor operator and the breaker is locked, sealed or otherwise secured in the off position:

Unit 1	Unit 2
MOV-1869A	MOV-2869A
MOV-1869B	MOV-2869B
MOV-1890A	MOV-2890A
MOV-1890B	MOV-2890B

c. Power may be restored to any valve or breaker referenced in Specifications 4.11.C.4.a and 4.11.C.4.b for the purpose of testing or maintenance provided that not more than one valve has power restored at one time, and the testing and maintenance is completed and power removed within 24 hours.

5. Verifying:

   a. That each automatic valve capable of receiving a safety injection signal, actuates to its correct position upon receipt of a safety injection test signal. The charging and low head safety injection pumps may be immobilized for this test.

   b. That each charging pump and safety injection pump circuit breaker actuates to its correct position upon receipt of a safety injection test signal. The charging and low head safety injection pumps may be immobilized for this test.

   c. By visual inspection that the low head safety injection containment sump components are not restricted by debris and show no evidence of structural distress or abnormal corrosion.

Basis

Complete system tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation. The method of assuring operability of these systems is therefore to combine system tests to be performed during unit outages, with more frequent component tests, which can be performed during reactor operation.

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.

Periodic inspections of containment sump components ensure that the components are unrestricted and stay in proper operating condition. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

<u>References</u>

UFSAR Section 6.2, Safety Injection System

4.12 AUXILIARY VENTILATION EXHAUST FILTER TRAINS

Applicability

Applies to the testing of safety-related air filtration systems.

Objective

To verify that leakage efficiency and iodine removal efficiency are within acceptable limits.

Specifications

A. Tests and Frequency

The following Tests shall be performed at the frequencies specified in the Surveillance Frequency Control Program or as specified below and as required for the conditions identified below:

1. Operate each redundant filter train circuit.

2. Demonstrate the operability of the entire safety-related portion of the auxiliary ventilation system.

3. Determine auxiliary ventilation system exhaust fan flow rate through each filter train in the LOCA mode of operation initially, after any structural maintenance on the HEPA filter or charcoal adsorber housings, once per 18 months, or after partial or complete replacement of the HEPA filters of charcoal adsorbers.

   The procedure for determining the air flow rate shall be in accordance with Section 9 of the ACGIH Industrial Ventilation document and Section 8 of ANSI N510-1975.

4. Conduct a visual inspection of the filter train and associated components before each in-place air flow distribution test, DOP test, or activated charcoal adsorber leak test in accordance with the intent of Section 5 of ANSI N510-1975.

5. Perform an air distribution test across the prefilter bank initially and after any major modification, major repair, or maintenance of the air cleaning system affecting the filter bank flow distribution. The air distribution test shall be performed with an anemometer located at the downstream side and at the center of each carbon filter.

6. Perform in-place cold DOP tests for HEPA filter banks:

   a. Initially;

   b. Once per 18 months;

   c. Following painting, fire, or chemical release in any ventilation zone communicating with the system during system operation;

   d. After each complete or partial replacement of the HEPA filter cells; and

   e. After any structural maintenance on the filter housing.

   The procedure for in-place cold DOP tests shall be in accordance with ANSI N510-1975, Section 10.5 or 11.4. The flow rate during the in-place cold DOP tests shall be 36,000 CFM ±10 percent. The flow rate shall be determined by recording the flow meter reading in the control room.

7. Perform in-place halogenated hydrocarbon leakage tests for the charcoal adsorber bank:

   a. Initially;

   b. Once per 18 months;

c. Following painting, fire, or chemical release in any ventilation zone communicating with the system during system operation;

d. After each complete or partial replacement of charcoal adsorber trays; and

e. After any structural maintenance of the filter housing.

The procedure for in-place halogenated hydrocarbon leakage tests shall be in accordance with ANSI N510-1975, Section 12.5. The flow rate during the in-place halogenated hydrocarbon leakage tests shall be 36,000 CFM ±10 percent. The flow rate shall be determined by recording the flow meter reading in the control room.

8. Perform laboratory analysis of each charcoal train:

a. Initially, whenever a new batch of charcoal is used to fill adsorbers trays; and

b. After 720 hours of train operation; and

c. Following painting, fire, or chemical release in any ventilation zone communicating with the system during system operation; and

d. After any structural maintenance on the HEPA filter or charcoal adsorber housings that could affect operation of the charcoal adsorber; and

e. At least once per eighteen months, if not otherwise performed per condition 8.b, 8.c, or 8.d within the last eighteen months.

The procedure for iodine removal efficiency tests shall follow ASTM D3803. The test conditions shall be in accordance with those listed in Specification 4.12.B.7.

9. Check the pressure drop across the HEPA filter and adsorber banks:

    a. Initially;

    b. Once per 18 months thereafter for systems maintained in a standby status and after 720 hours of system operation; and

    c. After each complete or partial replacement of filters or adsorbers.

B. <u>Acceptance Criteria</u>

1. The minimum period of air flow through the filters shall be 15 minutes.

2. The system operability test of Specification 4.12.A.2 shall demonstrate automatic start-up, shutdown and flow path alignment.

3. The air flow rate determined in Specification 4.12.A.3 shall be:

    a. 36,000 cfm ±10 percent with system in the LOCA mode of operation.

    b. The ventilation system shall be adjusted until the above limit is met.

4. Air distribution test across the prefilter-bank shall show uniformity of air velocity within ± 20 percent of average velocity. The ventilation system shall be adjusted until the limit is met.

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 at the frequency specified in the Surveillance Frequency Control Program prevents moisture buildup in the filters and adsorbers. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

4.13 RCS OPERATIONAL LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever T_{avg} (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Objective

To verify that RCS operational LEAKAGE is maintained within the allowable limits, the following surveillances shall be performed at the frequencies specified in the Surveillance Frequency Control Program.

Specifications

A. Verify RCS operational LEAKAGE is within the limits specified in TS 3.1.C by performance of RCS water inventory balance.[1, 2]

B. Verify primary to secondary LEAKAGE is ≤ 150 gallons per day 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.

Notes:

1. Not required to be completed until 12 hours after establishment of steady state operation.
2. Not applicable to primary to secondary LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.13.A

Verifying RCS LEAKAGE to be within the Limiting Condition for Operation (LCO) limits ensures the integrity of the reactor coolant pressure boundary (RCPB) is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The surveillance is modified by two notes. Note 1 states that this SR is not required to be completed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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 primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 4). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 4.13.A and SR 4.13.B / Note 1

With respect to SR 4.13.A and SR 4.13.B, as the associated Note 1 modifies the required completion of the surveillance, it is construed to be part of the specified completion time. Should the surveillance interval be exceeded while steady state operation has not been established, Note 1 allows 12 hours after establishment of steady state operation to complete the surveillance. The surveillance is still considered to be completed within the specified completion time. Therefore, if the surveillance were not completed within the required surveillance interval (plus extension allowed by TS 4.0.2) interval, but steady state operation had not been established, it would not constitute a failure of the SR. Once steady state operation is established, 12 hours would be allowed for completing the surveillance. If the surveillance were not completed within this 12 hour interval, there would a failure to complete a surveillance within the specified completion time, and the provisions of SR 4.0.3 would apply.

REFERENCES

1. UFSAR, Chapter 4, Surry Units 1 and 2.
2. UFSAR, Chapter 14, Surry Units 1 and 2.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B. Surveillance Requirements

1. Test for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the

   Commission or an agreement State as follows:

   a. Each sealed source, except startup sources subject to core flux, containing radioactive material other than Hydrogen 3 with a half-life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at the frequency specified in the Surveillance Frequency Control Program.

   b. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested at the frequency specified in the Surveillance Frequency Control Program prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within the frequency specified in the Surveillance Frequency Control Program prior to the transfer, sealed sources shall not be put into use until tested.

   c. Startup sources shall be leak tested prior to and following any repair or maintenance and before being subjected to core flux.

2. A complete inventory of radioactive materials in possession shall be maintained current at all times.

Basis

Ingestion or inhalation of source material may give rise to total body or organ irradiation. This specification assures that leakage from radioactive materials sources does not exceed allowable limits. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39(c) limits for plutonium.

4.18 MAIN CONTROL ROOM/EMERGENCY SWITCHGEAR ROOM (MCR/ESGR) EMERGENCY VENTILATION SYSTEM (EVS) TESTING

A. Operate each MCR/ESGR EVS train for ≥ 15 minutes in accordance with the frequency specified in the Surveillance Frequency Control Program.

B. Perform required Control Room Air Filtration System Testing in accordance with TS 4.20.

C. Perform required MCR/ESGR envelope unfiltered air inleakage testing in accordance with the MCR/ESGR Envelope Habitability Program.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.18.A

Standby systems should be checked periodically to ensure that they function properly. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Operation of the MCR/ESGR EVS trains shall be initiated manually from the MCR.

SR 4.18.B

This SR verifies that the required Control Room Air Filtration System testing is performed in accordance with Specification 4.20. Specification 4.20 includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in TS 4.20.

SR 4.18.C

This SR verifies the OPERABILITY of the MCR/ESGR envelope boundary by testing for unfiltered air inleakage past the MCR/ESGR envelope boundary and into the MCR/ESGR envelope. The details of the testing are specified in the MCR/ESGR Envelope Habitability Program (TS 6.4.R).

4. Measurement, at designated locations, of the MCR/ESGR envelope pressure relative to all external areas adjacent to the MCR/ESGR envelope boundary during the pressurization mode of operation by one train of the MCR/ESGR EVS, operating at the flow rate required by TS 4.20, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the assessment of the MCR/ESGR envelope boundary.
5. The quantitative limits on unfiltered air inleakage into the MCR/ESGR envelope. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph 3. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of MCR/ESGR envelope occupants to these hazards will be within the assumptions in the licensing basis.
6. The provisions of SR 4.0.2 are applicable to the Frequencies for assessing MCR/ESGR envelope habitability, determining MCR/ESGR envelope unfiltered inleakage, and measuring MCR/ESGR envelope pressure and assessing the MCR/ESGR envelope boundary as required by paragraphs 3 and 4, respectively.

S. Surveillance Frequency Control Program (SFCP)

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specification are performed at interval sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 273 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 272 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated March 30, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100900391), as supplemented by letters dated August 23, 2010 (ADAMS Accession No. ML102430164), and March 4, 2011 (ADAMS Accession No. ML110660230), Virginia Electric and Power Company (VEPCO, the licensee) submitted a request for changes to the Surry Power Station, Unit Nos. 1 and 2 (Surry Units 1 and 2), Technical Specifications (TSs). The supplements dated August 23, 2010, and March 4, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 10, 2010 (75 FR 48377).

The requested change is the adoption of the Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF-425), Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF [Risk-Informed Technical Specification Task Force] Initiative 5b" (Reference 1). When implemented, TSTF–425 relocates most periodic frequencies of TS surveillances to a licensee-controlled program, the Surveillance Frequency Control Program (SFCP), and provides requirements for the new program in the Administrative Controls section of the TS. All surveillance frequencies can be relocated except:

- Frequencies that reference other approved programs for the specific interval (such as the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program);
- Frequencies that are purely event-driven (e.g., "each time the control rod is withdrawn to the 'full out' position");

- Frequencies that are event-driven, but have a time component for performing the surveillance on a one-time basis once the event occurs (e.g., "within 24 hours after thermal power reaching ≥ 95 percent RTP [rated thermal power]"); and

- Frequencies that are related to specific conditions (e.g., battery degradation, age, and capacity) or conditions for the performance of a surveillance requirement (SR) (e.g., "drywell to suppression chamber differential pressure decrease").

A new program is added to the Administrative Controls of TS Section 5 as Specification 5.5.16. The new program is called the "Surveillance Frequency Control Program (SFCP)," and describes the requirements for the program to control changes to the relocated surveillance frequencies. The TS Bases for each of the affected SRs are revised to state that the frequency is set in accordance with the SFCP. Some SRs Bases do not contain a discussion of the frequency. In these cases, the Bases describing the current frequency were added to maintain consistency with the Bases for similar surveillances. These instances are noted in the markup along with the source of the text. The proposed licensee changes to the Administrative Controls of the TS to incorporate the SFCP include a specific reference to Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1 (Reference 2), as the basis for making any changes to the surveillance frequencies once they are relocated out of the TS.

In a letter dated September 19, 2007, the NRC staff approved NEI 04–10, Revision 1 (ADAMS Accession No. ML072570267), as acceptable for referencing in licensing actions to the extent specified and under the limitations delineated in NEI 04–10, and the safety evaluation providing the basis for NRC acceptance of NEI 04–10.

2.0 REGULATORY EVALUATION

In the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants," published in the *Federal Register* (*FR*) (58 *FR* 39132-39139, July 22, 1993), the NRC addressed the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Assessment or PRA) in Standard Technical Specifications. In discussing the use of PSA in Nuclear Power Plant Technical Specifications, the Commission wrote in part:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that technical specifications can be relaxed or removed, a deterministic review will be performed. ....
>
> The Commission Policy in this regard is consistent with its Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "* * * probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made * * * about the degree of confidence to be given these [probabilistic] estimates and assumptions. This is a key part of the process of determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety." ....

> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification change.

58 *Federal Register* at 39135 (alteration in original)

Approximately 2 years later, the NRC provided additional detail concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities" published in the *Federal Register* (60 *FR* 42622, August 16, 1995). The Commission, in discussing the deterministic and probabilistic approach to regulation, and the Commission's extension and enhancement of traditional regulation, wrote in part:

> PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner.

The Commission provided its new policy, stating:

> Although PRA methods and information have thus far been used successfully in nuclear regulatory activities, there have been concerns that PRA methods are not consistently applied throughout the agency, that sufficient agency PRA/statistics expertise is not available, and that the Commission is not deriving full benefit from the large agency and industry investment in the developed risk assessment methods. Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data. Implementation of the policy statement will improve the regulatory process in three areas: Foremost, through safety decision making enhanced by the use of PRA insights; through more efficient use of agency resources; and through a reduction in unnecessary burdens on licensees.
>
> Therefore, the Commission adopts the following policy statement regarding the expanded NRC use of PRA:
>
> (1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
>
> (2) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism

associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

(3) PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

(4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

As stated in 10 CFR 50.36(c)(3), *"Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."* These categories will remain in the TSs. The new TS SFCP provides the necessary administrative controls to require that surveillances relocated to the SFCP are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04–10, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of structures, systems, and components (SSCs), and are required to be documented. Furthermore, changes to frequencies are subject to regulatory review and oversight of the SFCP implementation through the rigorous NRC review of safety-related SSC performance provided by the reactor oversight program (ROP).

Licensees are required by the TSs to perform surveillance test, calibration, or inspection on specific safety-related system equipment (e.g., reactivity control, power distribution, electrical, and instrumentation) to verify system operability. Surveillance frequencies, currently identified in the TSs, are based primarily upon deterministic methods such as engineering judgment, operating experience, and manufacturer's recommendations. The licensee's use of NRC-approved methodologies identified in NEI 04–10 (Reference 2) provides a way to establish risk-informed surveillance frequencies that complement the deterministic approach and support the NRC's traditional defense-in-depth philosophy.

The licensee's SFCP ensures that surveillance requirements specified in the TSs are performed at intervals sufficient to assure the above regulatory requirements are met. Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and 10 CFR, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (corrective action program), require licensee monitoring of surveillance test failures and implementing corrective actions to address such failures. One of these actions may be to consider increasing the frequency at which a surveillance test is performed. In addition, the SFCP implementation guidance in NEI 04–10 requires monitoring the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs. These requirements, and the monitoring required by NEI 04–10, ensure that surveillance frequencies are sufficient to assure that the requirements of 10 CFR 50.36 are satisfied and that any performance deficiencies will be identified and appropriate corrective actions taken.

Regulatory Guide (RG) 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 5), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (Reference 3), describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 4), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors.

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF–425 for Surry Units 1 and 2 provides for administrative relocation of applicable surveillance frequencies, and provides for the addition of the SFCP to the administrative controls of the TSs. TSTF–425 also requires the application of NEI 04–10 for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes proposed in TSTF–425 included documentation regarding the PRA technical adequacy consistent with the requirements of RG 1.200. In accordance with NEI 04–10, PRA methods are used, in combination with plant performance data and other considerations, to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is in accordance with guidance provided in RG 1.174 and RG 1.177 in support of changes to surveillance test intervals.

3.1 RG 1.177, Five Key Safety Principles

RG 1.177 identifies five key safety principles required for risk-informed changes to the TSs. Each of these principles is addressed by the industry methodology document, NEI 04–10.

3.1.1 The Proposed Change Meets Current Regulations

Paragraph 50.36(c)(3) provides that the TSs will include surveillances which are "requirements relating to test, calibration, or inspection to assure that necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." NEI 04-10 provides guidance for relocating the surveillance frequencies from the TSs to a licensee-controlled program by providing an NRC-approved methodology for control of the surveillance frequencies. The surveillances themselves would remain in the TSs, as required by 10 CFR 50.36(c)(3).

This change is consistent with other NRC-approved TS changes in which the surveillance frequencies are relocated to licensee-controlled documents, such as surveillances performed in accordance with the In-service Testing Program or the Primary Containment Leakage Rate Testing Program. Thus, this proposed change meets the first key safety principle of RG 1.177 by complying with current regulations.

3.1.2 The Proposed Change Is Consistent With the Defense-in-Depth Philosophy

Consistency with the defense-in-depth philosophy, the second key safety principle of RG 1.177, is maintained if:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers). Because the scope of the proposed methodology is limited to revision of surveillance frequencies, the redundancy, independence, and diversity of plant systems are not impacted.
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A, is maintained.

TSTF–425 requires the application of NEI 04–10 for any changes to surveillance frequencies within the SFCP. NEI 04–10 uses both the core damage frequency (CDF) and the large early release frequency (LERF) metrics to evaluate the impact of proposed changes to surveillance frequencies. The guidance of RG 1.174 and RG 1.177 for changes to the CDF and the LERF is achieved by evaluation using a comprehensive risk analysis, which assesses the impact of proposed changes including contributions from human errors and common cause failures.

Defense-in-depth is also included in the methodology explicitly as a qualitative consideration outside of the risk analysis, as is the potential impact on detection of component degradation that could lead to an increased likelihood of common cause failures. Both the quantitative risk analysis and the qualitative considerations assure a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177.

3.1.3 The Proposed Change Maintains Sufficient Safety Margins

The engineering evaluation that will be conducted by the licensee under the SFCP when frequencies are revised will assess the impact of the proposed frequency change with the principle that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the proposed surveillance test frequency change is not in conflict with approved industry codes and standards or adversely affects any assumptions or inputs to the safety analysis, or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

The design, operation, testing methods, and acceptance criteria for SSCs, specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the Updated Final Safety Analysis Report and bases to the TSs), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. Thus, safety margins are maintained by the proposed methodology, and the third key safety principle of RG 1.177 is satisfied.

3.1.4 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 (Reference 12).

RG 1.177 provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies. This requires the identification of the risk contribution from impacted surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations. TSTF–425 requires application of NEI 04–10 in the SFCP. NEI 04–10 satisfies the intent of RG 1.177 requirements for evaluating the change in risk, and for assuring that such changes are small.

3.1.4.1 Quality of the PRA

The quality of the Surry Units 1 and 2 PRA is compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change. That is, the more the potential change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the quality of the PRA.

The licensee used RG 1.200 to address the technical adequacy of the Surry Units 1 and 2 PRA. RG 1.200 is NRC's developed regulatory guidance, which endorses with comments and qualifications the use of the American Society of Mechanical Engineers (ASME) RA–Sb–2005, "Addenda B to ASME RA–S–2002, 'Standard for Probabilistic Risk Assessment for Nuclear Power Plant Application'" (Reference 6), NEI 00–02, Revision 1, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance" (Reference 7), and NEI 05–04, Revision 0, "Process for

Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard" (Reference 8). The licensee has performed an assessment of the PRA models used to support the SFCP using the guidance of RG 1.200 to assure that the PRA models are capable of determining the change in risk due to changes to surveillance frequencies of SSCs, using plant-specific data and models. Capability category II of ASME RA–Sb–2005 was applied as the standard, and any identified deficiencies to those requirements are assessed further to determine any impacts to proposed decreases to surveillance frequencies, including by the use of sensitivity studies where appropriate.

The original industry peer review of the Surry Units 1 and 2 internal events PRA was conducted in 1998. The PRA model has been updated to address all findings from this review except for three documentation issues which remain open. These do not impact the PRA results and do not affect the technical adequacy of the PRA model.

The licensee performed a self-assessment of the PRA to supplement the peer review and assure all elements of the PRA standard are assessed (a "gap" assessment). This is consistent with RG 1.200, and is one acceptable method to determine conformance with the standard. In addition, a focused scope peer review was completed in February 2010, and the results of this assessment were provided to the staff in the October 26, 2010, supplemental submittal (Reference 11). The staff reviewed the results of the licensee's assessment of the Surry Units 1 and 2 PRA and the focused scope peer review, and the remaining open deficiencies that do not conform to capability category II of the ASME PRA standard (Table 1 of the October 26, 2010, supplement). These reviews identified 12 "gaps" to the requirements of the standard. Seven of the items were related to documentation deficiencies which do not impact the technical adequacy of the model. The remaining five technical deficiencies are discussed below.

Gap #2 and #11: For internal flooding initiators, the potential impacts of jet impingement, pipe whip, humidity, and other types of failures are not documented. The licensee stated that walkdown sheets will be enhanced to better capture spatial impacts, and that sensitivity evaluation will be used when necessary for additional flood impacts. These additional failure modes may be addressed and dispositioned for each surveillance frequency evaluation per the NEI 04–10 methodology.

Gap #9: For support system initiators evaluated by fault tree, the focus scope peer review identified that 1) not all possible combinations of cutsets are captured, and 2) there is no comparison with generic data sources for these initiators. The licensee will revise the initiator fault tree models consistent with current industry guidance when evaluating any surveillance interval changes associated with affected systems in a sensitivity model. Therefore, this deficiency may be addressed and dispositioned for each surveillance frequency evaluation per the NEI 04–10 methodology.

Gap #10: The licensee used a human reliability analysis methodology which is not sufficient to meet the standard. The human error probabilities will be recalculated using an appropriate method in a sensitivity model. Therefore, this deficiency may be addressed and dispositioned for each surveillance frequency evaluation per the NEI 04–10 methodology.

Gap #12: Limitation of the quantification process are not documented. The licensee stated that this will be addressed for each surveillance frequency evaluation, and so this deficiency may be addressed and dispositioned for each surveillance frequency evaluation per the NEI 04–10 methodology.

Based on the licensee's assessment using the applicable PRA standard and RG 1.200, the level of PRA quality, combined with the proposed evaluation and disposition of gaps described above, is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP, and is consistent with Regulatory Position 2.3.1 of RG 1.177.

3.1.4.2 Scope of the PRA

The licensee is required to evaluate each proposed change to a relocated surveillance frequency using the guidance contained in NEI 04–10 to determine its potential impact on risk, due to impacts from internal events, fires, seismic, other external events, and from shutdown conditions. Consideration is made of both the CDF and the LERF metrics. In cases where a PRA of sufficient scope or where quantitative risk models were unavailable, the licensee uses bounding analyses, or other conservative quantitative evaluations. A qualitative screening analysis may be used when the surveillance frequency impact on plant risk is shown to be negligible or zero.

The individual plant examination of external events (IPEEE) fire-induced vulnerability evaluation analysis and fire PRA, and the IPEEE focused scope seismic PRA, will be used to provide insights for fires and seismic events. Other external hazards will be assessed using a qualitative or bounding approach for this application.

The licensee's evaluation methodology is sufficient to ensure the scope of the risk contribution of each surveillance frequency change is properly identified for evaluation, and is consistent with Regulatory Position 2.3.2 of RG 1.177.

3.1.4.3 PRA Modeling

The licensee will determine whether the SSCs affected by a proposed change to a surveillance frequency are modeled in the PRA. Where the SSC is directly or implicitly modeled, a quantitative evaluation of the risk impact may be carried out. The methodology adjusts the failure probability of the impacted SSCs, including any impacted common cause failure modes, based on the proposed change to the surveillance frequency. Where the SSC is not modeled in the PRA, bounding analyses are performed to characterize the impact of the proposed change to the surveillance frequency. Potential impacts on the risk analyses due to screening criteria and truncation levels are addressed by the requirements for PRA technical adequacy consistent with guidance contained in RG 1.200, and by sensitivity studies identified in NEI 04–10.

The licensee will perform quantitative evaluations of the impact of selected testing strategy (i.e., staggered testing or sequential testing) consistently with the guidance of NUREG/CR–6141 and NUREG/CR–5497, as discussed in NEI 04–10.

Thus, through the application of NEI 04–10, the Surry Units 1 and 2 PRA modeling is sufficient to ensure an acceptable evaluation of risk for the proposed changes in surveillance frequency, and is consistent with Regulatory Position 2.3.3 of RG 1.177.

3.1.4.4 Assumptions for Time Related Failure Contributions

The failure probabilities of SSCs modeled in the Surry Units 1 and 2 PRA include a standby time-related contribution and a cyclic demand-related contribution. NEI 04–10 criteria adjust the time-related failure contribution of SSCs affected by the proposed change to surveillance frequency. This is consistent with RG 1.177, Section 2.3.3 which permits separation of the failure rate contributions into demand and standby for evaluation of surveillance requirements. If the available data do not support distinguishing between the time-related failures and demand failures, then the change to surveillance frequency is conservatively assumed to impact the total failure probability of the SSC, including both standby and demand contributions. The SSC failure rate (per unit time) is assumed to be unaffected by the change in test frequency, and will be confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented. The process requires consideration of qualitative sources of information with regards to potential impacts of test frequency on SSC performance, including industry and plant-specific operating experience, vendor recommendations, industry standards, and code-specified test intervals. Thus the process is not reliant upon risk analyses as the sole basis for the proposed changes.

The potential beneficial risk impacts of reduced surveillance frequency, including reduced downtime, lesser potential for restoration errors, reduction of potential for test-caused transients, and reduced test-caused wear of equipment, are identified qualitatively, but are conservatively not required to be quantitatively assessed. Thus, through the application of NEI 04–10, the licensee has employed reasonable assumptions with regard to extensions of surveillance test intervals, and is consistent with Regulatory Position 2.3.4 of RG 1.177.

3.1.4.5 Sensitivity and Uncertainty Analyses

NEI 04–10 requires sensitivity studies to assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact to the frequency of initiating events, and of any identified deviations from capability category II of ASME PRA Standard (ASME RA–Sb–2005) (Reference 6). Where the sensitivity analyses identify a potential impact on the proposed change, revised surveillance frequencies are considered, along with any qualitative considerations that may bear on the results of such sensitivity studies. Required monitoring and feedback of SSC performance once the revised surveillance frequencies are implemented will also be performed. Thus, through the application of NEI 04–10, the licensee has appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations, and is consistent with Regulatory Position 2.3.5 of RG 1.177.

3.1.4.6 Acceptance Guidelines

The licensee will quantitatively evaluate the change in total risk (including internal and external events contributions) in terms of the CDF and the LERF for both the individual risk impact of a proposed change in surveillance frequency and the cumulative impact from all individual changes to surveillance frequencies using the guidance contained in NRC-approved NEI 04–10 in accordance with the TS SFCP. Each individual change to surveillance frequency must show a risk impact below 1E–6 per year for a change to the CDF, and below 1E–7 per year for a change to the LERF. These are consistent with the limits of RG 1.174 for very small changes in risk. Where the RG 1.174 limits are not met, the process either considers revised surveillance frequencies which are consistent with RG 1.174 or the process terminates without permitting the proposed changes. Where quantitative results are unavailable to permit comparison to acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to surveillance frequency is negligible or zero. Otherwise, bounding quantitative analyses are required which demonstrate the risk impact is at least one order of magnitude lower than the RG 1.174 acceptance guidelines for very small changes in risk. In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk impact below 1E–5 per year for a change to the CDF, and below 1E–6 per year for a change to the LERF, and the total CDF and total LERF must be reasonably shown to be less than 1E–4 per year and 1E–5 per year, respectively. These are consistent with the limits of RG 1.174 for acceptable changes in risk, as referenced by RG 1.177 for changes to surveillance frequencies. The staff interprets this assessment of cumulative risk as a requirement to calculate the change in risk from a baseline model utilizing failure probabilities based on the surveillance frequencies prior to implementation of the SFCP, compared to a revised model with failure probabilities based on changed surveillance frequencies. The staff further notes that the licensee includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with insignificant risk increases (less than 5E–8 CDF and 5E–9 LERF) once the baseline PRA models are updated to include the effects of the revised surveillance frequencies.

The quantitative acceptance guidance of RG 1.174 is supplemented by qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, industry standards, the results of sensitivity studies, and SSC performance data and test history.

The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results compared to numerical acceptance guidelines. Post implementation performance monitoring and feedback are also required to assure continued reliability of the components. The licensee's application of NEI 04–10 provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177. Therefore, the proposed methodology satisfies the fourth key safety principle of RG 1.177 by assuring any increase in risk is small consistent with the intent of the Commission's Safety Goal Policy Statement (Reference 12).

3.1.5 The Impact of the Proposed Change Should Be Monitored Using Performance Measurement Strategies

The licensee's adoption of TSTF–425 requires application of NEI 04–10 in the SFCP. NEI 04–10 requires performance monitoring of SSCs whose surveillance frequency has been revised as part of a feedback process to assure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback includes consideration of maintenance rule monitoring of equipment performance. In the event of degradation of SSC performance, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions which may apply as part of the maintenance rule requirements. The performance monitoring and feedback specified in NEI 04–10 is sufficient to reasonably assure acceptable SSC performance and is consistent with Regulatory Position 3.2 of RG 1.177. Thus, the fifth key safety principle of RG 1.177 is satisfied.

3.2 Addition of Surveillance Frequency Control Program to Administrative Controls

The licensee has included the SFCP and specific requirements into the Administrative Controls, TS Section 6.4.S, Surveillance Frequency Control Program, as follows:

> This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specification are performed at intervals sufficient to assure that the associated Limiting Conditions for Operation are met.
>
> a. The Surveillance Frequency Control Program shall contain a list of Frequencies of the Surveillance Requirements for which the Frequency is controlled by the program.
>
> b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04–10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
>
> c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

The proposed program is consistent with the model application of TSTF-425, and is therefore acceptable.

3.3 Summary and Conclusions

The staff has reviewed the licensee's proposed relocation of some surveillance frequencies to a licensee controlled document, and controlling changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the administrative controls of the TSs. The SFCP and TS Section 6.4.S references NEI 04–10, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from the TSs to a licensee-controlled document, provided those frequencies are changed in accordance with NEI 04–10 which is specified in the Administrative Controls of the TSs.

The proposed licensee adoption of TSTF–425 and risk-informed methodology of NEI 04–10 as referenced in the Administrative Controls of the TSs, satisfies the key principles of risk-informed decision making applied to changes to the TS as delineated in RG 1.177 and RG 1.174, in that:

- The proposed change meets current regulations;
- The proposed change is consistent with defense-in-depth philosophy;
- The proposed change maintains sufficient safety margins;
- Increases in risk resulting from the proposed change are small and consistent with the Commission's Safety Goal Policy Statement; and

- The impact of the proposed change is monitored with performance measurement strategies.

Paragraph 50.36(c)(3) states "Surveillance Requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The NRC staff finds that with the proposed relocation of surveillance frequencies to an owner-controlled document and administratively controlled in accordance with the TS SFCP, VEPCO continues to meet the regulatory requirement of 10 CFR 50.36, and specifically, 10 CFR 50.36(c)(3), surveillance requirements.

The NRC has concluded, on the basis of the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (August 10, 2010, 75 FR 48377). The amendments also relate to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusions set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. TSTF–425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," March 18, 2009 (ADAMS Accession No. ML090850642).

2. NEI 04–10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," April 2007 (ADAMS Accession No. ML071360456).

3. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," NRC, August 1998 (ADAMS Accession No. ML003740176).

4. Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, January 2007 (ADAMS Accession No. ML070240001).

5. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," NRC, November 2002 (ADAMS Accession No. ML023240437).

6. ASME PRA Standard ASME RA–Sb–2005, "Addenda B to ASME RA–S–2002, 'Standard for Probabilistic Risk Assessment for Nuclear Power Plant Application.'"

7. NEI 00–02, Revision 1 "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," May 2006 (ADAMS Accession No. ML061510621).

8. NEI 05–04, Revision 0, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard, August 2006.

9. J. Alan Price, VEPCO to NRC, "Proposed License Amendment Request Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," March 30, 2010 (ADAMS Accession No. ML100900391).

10. Leslie N. Hartz, VEPCO to NRC, "Supplemental Information for Proposed License Amendment Request Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," August 23, 2010 (ADAMS Accession No. ML102430164).

11. J. Alan Price, VEPCO to NRC, "Response to Request for Additional Information, License Amendment Request Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," October 26, 2010 (ADAMS Accession No. ML102990293).

12. "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement; Correction and Republication" (51 FR 30028, August 21, 1986; "Safety Goal Policy Statement").

Principal Contributor: Andrew Howe, NRR/DRA

Date: April 29, 2011

April 29, 2011

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING RELOCATION OF SURVEILLANCE FREQUENCIES TO LICENSEE-CONTROLLED PROGRAM USING RISK-INFORMED JUSTIFICATION (TSTF-425) (TAC NOS. ME3687 AND ME3688)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 273 to Renewed Facility Operating License No. DPR-32 and Amendment No. 272 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments revise the Technical Specifications (TSs) in response to your application dated March 30, 2010, as supplemented by letters dated August 23, 2010, and March 4, 2011.

These amendments revise the TSs by relocating specific surveillance frequencies to a licensee-controlled document using a risk-informed justification.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA by JStang for/

Karen Cotton, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 273 to DPR-32
2. Amendment No. 272 to DPR-37
3. Safety Evaluation

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ADAMS Accession No. ML110740033 ***SE transmitted by memo dated**

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	DRA/APLA/BC	NRR/ITSB/BC	OGC/NLO	NRR/LPL2-1/BC	NRR/LPL2-1/PM
NAME	KCotton	MO'Brien (JBurkhardt for)	DHarrison*	RElliott	BHarris w/ comment	GKulesa	KCotton (JStang for)
DATE	04/18/11	04/19/11	11/02/10	4/29/11	04/21/11	04/28/11	04/29/11

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