June 19, 1989

Docket Nos. 50-280
and 50-281

DISTRIBUTION
See attached sheet

Mr. W. R. Cartwright
Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Cartwright:

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: SERVICE WATER SYSTEM (TAC NOS. 72811 AND 72812)

The Commission has issued the enclosed Amendment No. 130 to Facility Operating License No. DPR-32 and Amendment No. 130 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated March 27, 1989.

These amendments revise the Technical Specifications relating to the canal level, canal safety-related level actuation instrumentation and the emergency service water pumps.

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,

Original signed by

Bart C. Buckley, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 130 to DPR-32
2. Amendment No. 130 to DPR-37
3. Safety Evaluation

cc w/enclosures:
See next page

LA:PDII-2	PM:PDII-2	D:PDII-2	OGC
DMiller	BBuckley/jd	HBerkow	
5/15/89	5/15/89	5/17/89	6/08/89

CP1 cc

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Virginia Electric and Power Company

Surry Power Station

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AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-32 - SURRY UNIT 1
AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-37 - SURRY UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
B. Buckley
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
T. Meek (8), P1-137
Wanda Jones, P-130A
E. Butcher, 11/F/23
W. LeFave, 8/D/1
ACRS (10)
GPA/PA
ARM/LFMB
P. Frederickson, R-II
Others as required

cc: Plant Service list

DF01
1/1

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
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 27, 1989,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 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. 130 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 19, 1989

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
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 27, 1989, 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 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. 130 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 19, 1989

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 130 FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 130 FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS 3.7-9b	TS 3.7-9b
TS 3.7-16	TS 3.7-16
TS 3.7-19	TS 3.7-19
TS 3.14-1	TS 3.14-1
TS 3.14-2	TS 3.14-2
TS 3.14-3	TS 3.14-3
TS 3.14-4	TS 3.14-4
TS 4.1-8a	TS 4.1-8a
TS 4.1-8aa	TS 4.1-8aa
TS 4.1-9c	TS 4.1-9c

control room.

The supply lines installed from the containment penetrations to the hydrogen analyzers have Category I Class IE heat tracing applied. The heat tracing system receives the same transferable emergency power as is provided to the containment hydrogen analyzers. The heat trace system is de-energized during normal system operation. Upon receipt of a safety injection signal (Train A or Train B), the system is automatically started, after a preset time delay, to bring the piping process temperature to 250°F ± 10°F within 20 minutes. Each heat trace circuit is equipped with an RTD to provide individual circuit readout, over temperature alarm and cycles the circuit to maintain the process temperatures via the solid state control modules.

The hydrogen analyzer heat trace system is equipped with high temperature, loss of D.C. power, loss of A.C. power, loss of control power and failure of automatic initiation alarms.

Control Room Chlorine Detection System

The operability of the chlorine detection system ensures that sufficient capability is available to promptly detect and automatically initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel, and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

Non-Essential Service Water Isolation System

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

TABLE 3.7-2

ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 DEGREE OF REDUN-DANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
d. Station Blackout Start Motor Driven Pump	2	0		Restore inoperable channel within 48 hours or be in hot shutdown within next 6 hours and in cold shutdown within the following 30 hours.
e. Trip of Main Feedwater Pumps Start Motor Pumps	1/Pump	1/Pump		Restore inoperable channel within 48 hours or be in hot shutdown within next 6 hours and in cold shutdown within the following 30 hours.
4. LOSS OF POWER				
a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	2/Bus	1/Bus		Place inoperable channel in tripped condition within one hour.
b. 4.16 KV Emergency Bus Undervoltage (Grid Degraded Voltage)	2/Bus	1/Bus		Place inoperable channel in tripped condition within one hour.
5. NON-ESSENTIAL SERVICE WATER ISOLATION				
a. Low Intake Canal Level	3	1	One train may be blocked for a period not to exceed two hours for the purpose of logic testing.	Place inoperable channel(s) in tripped condition within one hour.

Amendment Nos. 130 and 130

TABLE 3.7-4

ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
6.	AUXILIARY FEEDWATER		
	a. Steam Generator Water Level Low-Low	Aux. Feedwater Initiation S/G Blowdown Isolation	$\geq$ 5% narrow range
	b. RCP Undervoltage	Aux. Feedwater Initiation	$\geq$ 70% nominal
	c. Safety Injection	Aux. Feedwater Initiation	All S.I. setpoints
	d. Station Blackout	Aux. Feedwater Initiation	$\geq$ 46.7% nominal
	e. Main Feedwater Pump Trip	Aux. Feedwater Initiation	N.A.
7.	LOSS OF POWER		
	a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	Emergency Bus Separation and Diesel start	75^{+1}_{-1}% volts with a 2+5, -0.1 second time delay
	b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	Emergency Bus Separation and Diesel start	$90^{+1.0}_{-1.0}$% volts with a 60$\pm$3.0 second time delay (Non CLS, Non SI) 7$\pm$.35 second time delay (CLS or SI Conditions)
8.	NON-ESSENTIAL SERVICE WATER ISOLATION		
	a. Low Intake Canal Level	Isolation of Service water flow to non-essential loads	23 feet-6 inches

Amendment Nos. 130 and 130

3.14 CIRCULATING AND SERVICE WATER SYSTEMS

Applicability

Applies to the operational status of the Circulating and Service Water Systems.

Objective

To define those limiting conditions of the Circulating and Service Water Systems necessary to assure safe station operation.

Specification

A. The Reactor Coolant System temperature or pressure of a reactor unit shall not exceed 350°F or 450 psig, respectively, or the reactor shall not be critical unless:

1. The high level intake canal is filled to at least elevation +23.0 feet at the high level intake structure.

2. Unit subsystems, including piping and valves, shall be operable to the extent of being able to establish the following:

   a. Flow to and from one bearing cooling water heat exchanger.

   b. Flow to and from the component cooling heat exchangers required by Specification 3.13.

3. At least two circulating water pumps are operating or are operable.

4. Three emergency service water pumps are operable; these pumps will service both units simultaneously.

5. Two service water flowpaths to the charging pump service water subsystem are operable.

6. Two service water flowpaths to the recirculation spray subsystems are operable.

B. The requirements of Specification 3.14.A.4 may be modified to allow one Emergency Service Water pump to remain inoperable for a period not to exceed 7 days. If this pump is not operable in 7 days, then place both units in Hot Shutdown within the next 6 hours and Cold Shutdown within the next 30 hours.

The requirements of 3.14.A.4 may be modified to have two Emergency Service Water pumps operable with one unit in Cold Shutdown with combined Spent Fuel pit and shutdown unit decay heat loads of 25 million BTU/HR or less. One of the two remaining pumps may be inoperable for a period not to exceed 7 days. If this pump is not operable in 7 days, then place the operating unit in Hot Shutdown within the next 6 hours and Cold Shutdown within the next 30 hours.

C. There shall be an operating service water flow path to and from one operating main control and emergency switchgear rooms air conditioning condenser and at least one operable service water flow path to and from at least one operable main control and emergency switchgear rooms air conditioning condenser whenever fuel is loaded in reactor core. Refer to Section 3.23.C for air conditioning system operability requirements above cold shutdown.

D. The requirements of Specifications A-5 and A-6 may be modified to allow unit operation with only one operable flow path to the charging pump service water subsystem and to the recirculation spray subsystems. If the affected systems are not restored to the requirements of Specifications A-5 and A-6 within 24 hours,

the reactor shall be placed in a hot shutdown condition. If the requirements of Specifications A-5 and A-6 are not met within an additional 48 hours, the reactor shall be placed in a cold shutdown condition.

Basis

The Circulating and Service Water System are designed for the removal of heat resulting from the operation of various systems and components of either or both of the units. Untreated water, supplied from the James River and stored in the high level intake canal is circulated by gravity through the recirculation spray coolers and the bearing cooling water heat exchangers and to the charging pumps lubricating oil cooler service water pumps which supply service water to the charging pump lube oil coolers.

In addition, the Circulating and Service Water Systems supply cooling water to the component cooling water heat exchangers and to the main control and emergency switchgear rooms air conditioning condensers. The Component Cooling heat exchangers are used during normal plant operations to cool various station components and when in shutdown to remove residual heat from the reactor. Component Cooling is not required on the accident unit during a loss-of-coolant accident. If the loss-of-coolant accident is coincident with a loss of off-site power, the nonaccident unit will be maintained at Hot Shutdown with the ability to reach Cold Shutdown.

The long term Service Water requirement for a loss-of-coolant accident in one unit with simultaneous loss-of-station power and the second unit being brought to Hot Shutdown is greater than 15,000 gpm. Additional Service Water is necessary to bring the nonaccident unit to Cold Shutdown. Three diesel driven Emergency Service Water pumps with a design capacity of 15,000 gpm each, are provided to supply water to the High Level Intake canal during a loss-of-station power incident. Thus, considering the single active failure of one pump, three Emergency Service Water pumps are required to be operable. The allowed outage time of 7 days provides operational flexibility to allow for repairs up to and

including replacement of an Emergency Service Water pump without forcing dual unit outages, yet limits the amount of operating time without the specified number of pumps.

When one Unit is in Cold Shutdown and the heat load from the shutdown unit and spent fuel pool drops to less than 25 million BTU/HR, then one Emergency Service Water pump may be removed from service for the subsequent time that the unit remains in Cold Shutdown due to the reduced residual heat removal and hence component cooling requirements.

A minimum level of +17.2 feet in the High Level Intake canal is required to provide design flow of Service Water through the Recirculation Spray heat exchangers during a loss-of-coolant accident for the first 24 hours. If the water level falls below +23'6", signals are generated to trip both unit's turbines and to close the nonessential Circulating and Service Water valves. A High Level Intake canal level of +23'6" ensures actuation prior to canal level falling to elevation +23'. The Circulating Water and Service Water isolation valves which are required to close to conserve Intake Canal inventory are periodically verified to limit total leakage flow out of the Intake Canal. In addition, passive vacuum breakers are installed on the Circulating Water pump discharge lines to assure that a reverse siphon is not continued for canal levels less than +23 feet when Circulating Water pumps are de-energized. The remaining six feet of canal level is provided coincident with ESW pump operation as the required source of Service Water for heat loads following the Design Basis Accident.

References:

FSAR Section 9.9	Service Water System
FSAR Section 10.3.4	Circulating Water System
FSAR Section 14.5	Loss-of-Coolant Accidents, Including the Design Basis Accident

TABLE 4.1-1
MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS, AND
TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
33.	Loss of Power				
	a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	M	
	b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	M	
34.	Control Room Chlorine Detectors	S.	R	M	
35.	Manual Reactor Trip	N.A.	N.A.	R	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.	M(1), R(2)	(1) Local manual undervoltage trip prior to placing breaker in service. (2) Automatic shunt trip.
37.	Safety Injection Input from ESF	N.A.	N.A.	R	
38.	Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	

TABLE 4.1-1
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	S	R	M	
40.	Intake Canal Low (See Footnote 1)	D	R	M(1), Q(2)	(1) Logic Test (2) Channel Electronics Test

S - Each Shift
D - Daily
N.A. - Not Applicable
Q - Every 90 effective full power days

M - Monthly
P - Prior to each startup if not done within the previous week
R - Each Refueling Shutdown

* See Specification 4.1.D

Footnote 1:

Check — Consists of verifying for an indicated intake canal level greater than 23'-6" 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 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.

Test (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 4.1-2A (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	Each refueling	9.9
b.	Service Water System Valves Isolating Flow to Non-essential loads on Intake Canal Low Level Isolation	Functional	Refueling	9.9
15.	Control Room Ventilation System	*Ability to maintain positive pressure for 1 hour using a volume of air equivalent to or less than stored in the bottled air supply	Each refueling interval (approx. every 12-18 months)	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 $\leq$ 350°F and the Reactor Vessel Head is bolted	None
17.	Reactor Vessel Overpressure Mitigating System Backup Air Supply	Setpoint	Refueling	None

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO.130 TO 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

The service water (SW) system at Surry, Units 1 and 2, is unique when compared to the SW systems at most nuclear power plants. Rather than emergency service water (ESW) system pumps forcing flow through the component cooling water (CCW) system and other safety-related system heat exchangers, flow is induced via gravity feed from a high level intake canal (HLIC) which is required to have a certain level of water to provide adequate head in order to meet minimum flow requirements. However, the inventory in the intake canal is not sufficient to meet long-term cooling requirements, and, therefore, three diesel-driven ESW pumps (actually makeup pumps) are provided to replenish the canal inventory in an emergency. During normal operation, canal inventory is maintained by the circulating water system (CWS) pumps, which also act as makeup pumps. The CWS flow through the main condensers is also gravity flow from the intake canal. The ESW and CWS pumps take suction from the James River.

As a result of a Safety System Functional Inspection (SSFI) during the weeks of September 12 and September 26, 1988, the NRC staff identified several concerns regarding the operability of the SW system that were directly related to the unique Surry SW system design. Following the identification of these concerns, the licensee re-evaluated the design basis for the SW system, particularly with respect to HLIC inventory control and management following a postulated design basis accident (DBA). As a result of this re-evaluation, the licensee, by letter dated March 27, 1989, identified design modifications and proposed Technical Specification changes necessary to correct deficiencies found during the SSFI and followup re-evaluation. The following evaluation addresses the design modifications and proposed Technical Specification changes identified in the licensee's March 27, 1989 submittal.

2.0 EVALUATION

The licensee has proposed three modifications to the plant Technical Specifications. These changes will:

1. Raise the minimum required HLIC level from 18 feet to 23 feet,

2. Increase the requirement from two to three ESW pumps to be operable with provisions for limited duration maintenance outages, and

3. Provide operability and surveillance requirements for the new safety-related HLIC level actuation system.

The proposed Technical Specification (TS) changes are accompanied by design modifications which include new vacuum breakers to prevent reverse siphoning at the new, higher canal level (23 feet), addition of a safety-related canal low level actuation system, repowering the circulating water system valves to assure isolation in the event of an emergency diesel generator failure, installation of manual vacuum breaker valves on the discharge tunnel to break prime and hence conserve inventory, and installation of CCW heat exchanger SW flow instrumentation to allow throttling during a DBA.

The licensee's re-evaluation identified six basic deficiencies which precipitated the proposed TS changes, design modifications and associated changes to station procedures as described below.

The first deficiency was related to HLIC level drawdown during a DBA due to a single failure. The DBA is assumed to be a loss of coolant accident (LOCA) and a loss of off-site power (LOOP). Possible single failure problems included an ESW pump failure, emergency diesel generator failure, and a failure to close any one of the isolation valves to heat exchangers not essential for post-DBA heat removal. To resolve this issue, emergency operating procedures (EOPs) were revised for operation of ESW pumps and SW heat exchangers, and require manual confirmation/action for closing specific SW isolation valves. Design modifications were also made to repower the CWS isolation valves, as required, to at least one CWS inlet or outlet valve for each condenser waterbox in order to ensure isolation following any single failure. In addition, to allow time for operator action, the TS would be revised to raise the minimum intake canal level from 18 feet to 23 feet.

The second deficiency noted was reliance on nonsafety-related signals for the main condenser and SW isolation valves. To resolve this concern, a safety-related canal low level actuation system was installed to provide safety-related signals to the system isolation valves. TS changes were also proposed to add operability and surveillance requirements for the new safety-related instrumentation.

A third deficiency was the potential for inadequate ESW supply (makeup) during a DBA in one unit with cooldown of the other unit or residual heat removal (RHR) system operation in the nonaccident unit (within a certain time interval following shutdown based on decay heat production). To resolve this problem, TS changes were proposed to (1) require three ESW pumps to be operable with two units at power, and (2) require three ESW pumps to be operable with one unit at power when heat loads from the spent fuel pool and a shutdown unit are equal to or greater than 25 million BTU/hr. The present TS only require the operability of two ESW pumps.

A fourth deficiency was the potential for the CWS pump discharge lines to siphon back to the James River on loss of the CWS pumps. The present design contains a passive vacuum breaker designed to prevent siphoning below a canal level of 19 feet, which is consistent with the present specified level of

18 feet in the TS. However, in order to be consistent with the proposed TS level of 23 feet, new passive vacuum breakers have been installed to assure a reverse siphon will not be present for canal levels less than or equal to 23 feet. The new vacuum breaker is located just above the 23 foot level.

The fifth deficiency identified by the licensee was the inability of the SW system to meet the requirements of Appendix R to 10 CFR Part 50 regarding safe shutdown following a fire. The current Appendix R analysis allows for failure of the CWS isolation valves to close followed by operator action. However, a method for breaking condenser vacuum under full flow conditions is not addressed in the current design. The resolution of this deficiency involves the installation of vacuum breakers on the condenser water boxes. The vacuum breakers will have pneumatic remote and local capability. The remote control stations are required in the event of a fire in the turbine building, and are located in a fire area that is appropriately separated from the turbine building by a fire wall in accordance with Appendix R criteria.

The sixth and final major deficiency identified by the licensee involved CCW heat exchanger operability in a degraded condition. The SW system side of the heat exchangers is subject to significant microfouling and macrofouling phenomena. The possibility of reduced performance of the heat exchangers must be accounted for in the design basis by application of appropriate fouling factors to both the SW and CCW modes of the heat exchangers. The resolution of this concern involves a combination of design modifications, procedural changes and periodic testing as follows:

1. The installation of differential pressure instrumentation on the SW side of the heat exchangers.

2. The installation of discharge tunnel vacuum breakers to conserve HLIC inventory for use in the CCW heat exchangers. These are safety-related valves (manually operated) which may have to be operated within 4 hours after a DBA if the discharge tunnel self-priming action is not already broken.

3. The implementation of a periodic test to measure combined CWS and SW system isolation valve leakage and a periodic test to measure CCW heat exchanger operability. The leakage test is required to demonstrate that valve leakage (HLIC inventory loss) is less than that assumed in canal inventory analysis following a DBA. Due to the seasonal variations in heat exchanger fouling, the period for the CCW heat exchanger operability test must be established by first setting a fairly high testing frequency until a reduction in that frequency can be substantiated.

A number of other potential design and operating deficiencies were identified by the licensee, which were related to the ESW pumps and pumphouse. Appropriate actions have been taken by the licensee to correct these potential deficiencies. These actions include design modifications, maintenance and testing procedural changes and additional instrumentation. These changes and modifications are conservative in nature and should make the ESW pumps more reliable.

The licensee has proposed two TS changes to TS 3.14 "Circulating and Service Water Systems." The first would increase the minimum required HLIC level to support SW heat exchanger flow by allowing for automatic and operator action times to isolate nonessential SW system flowpaths. The second proposed change to TS 3.14 requires three operable ESW pumps for HLIC makeup for long-term accident mitigation. Both of these proposed changes are more conservative than the original TS and are required to meet the design basis of the SWS as reanalyzed by the licensee. The revised design basis includes the most limiting DBA coupled with any single active failure.

A third TS change revises both TS 3.7 and 4.1 related to the engineered safety features (ESF) instrumentation setpoints, action requirements and testing frequencies. These TS changes are required to be consistent with the design modifications which included a safety-related intake canal level actuation system. The setpoints are consistent with the revised design basis and proposed changes to TS 3.14. The action statements and testing frequencies are consistent with those for other ESF instrumentation in the Surry plant TS.

The licensee has also revised the Bases Section of the plant TS to be consistent with the proposed TS changes and design modifications.

The design modifications described above are considered by the staff to be safety improvements, and provide added assurance that adequate canal inventory will be available and maintained to cope with the worst-case DBA coupled with any single active failure. These modifications improve the SW system post-accident capability for both the short and long term.

3.0 SUMMARY

As described above, the staff has reviewed and evaluated the licensee's design modifications and associated proposed TS changes related to the SW system and HLIC management. Based on its review, the staff concludes that the SW system and intake canal design modifications will ensure that the revised design basis is satisfied and meet the following staff criteria:

1. General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," as it relates to protection of the cooling water system against earthquakes and tornadoes.

2. GDC 44, "Cooling Water," as it relates to providing adequate cooling requirements under both normal and accident conditions, suitable redundancy, and the capability to isolate nonessential systems, components or piping if required; and Regulatory Guide 1.27, "Ultimate Heat Sinks for Nuclear Power Plants," as it relates to provisions for a reliable long-term source of cooling water to mitigate an accident in one unit and ensure a safe cold shutdown of the other unit.

3. GDC 45, "Inspection of Cooling Water Systems," as it relates to provisions to permit inservice inspection of safety-related cooling water system components and equipment.

4. GDC 46, "Testing of Cooling Water Systems," as it relates to design provisions to permit operational functional testing of safety-related cooling water system components and equipment.

The staff further concludes that the proposed TS changes are more conservative than the current specifications and are consistent with the design modifications and revised design basis analyses. The staff, therefore, concludes that the design modifications and proposed TS changes are acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on June 19, 1989 (54 FR 25761). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of the amendments will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 19, 1989

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