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JUN 23 1981

RECEIVED JUN 26 1981 U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos. 50-280
and 50-281

Mr. J. H. Ferguson
Executive Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Ferguson:

The Commission has issued the enclosed Amendment No. 71 to Facility Operating License No. DPR-32 and Amendment No. 71 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 May 19, 1981.

The Unit 1 amendment revises the Technical Specifications to reflect changes as a result of modifications made to alleviate Net Positive Suction Head (NPSH) problems with the Low Head Safety Injection and Recirculation Spray Pumps and modifications made to the containment spray system. Changes have been made to service water temperature, containment temperature, containment air partial pressure, refueling water storage tank volume and outside recirculation pump flow rate. These limits have been transferred to the Technical Specifications from the license.

The Unit 1 changes are the same as the changes made for Unit 2 in Amendment No. 59 on August 1, 1980, and therefore the Safety Evaluation (SE) supporting the Unit 2 changes applies to Unit 1 also. The August 1, 1980 SE states this. Therefore, we conclude that these changes are acceptable.

The Unit 2 amendment revises the Technical Specifications to reflect that the Unit 1 change above applies to both Units 1 and 2.

In preparing these amendments, we discovered that Amendment Nos. 66 and 65 dated February 25, 1981 inadvertently deleted paragraph 3.F from the licenses of Units 1 and 2. Paragraph 3.F was already deleted for Unit 2 by Amendment 59 dated August 1, 1980. This current action now correctly deletes Paragraph 3.F for Unit 1.

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OFFICE ▶							
SURNAME ▶							
DATE ▶							

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

We have concluded, based on the considerations discussed above, that; (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 71 to DPR-32
2. Amendment No. 71 to DPR-37
3. Notice of Issuance

cc: w/enclosures
See next page

No legal objection to issuance of amendment & notice. SER not reviewed.

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD		
SURNAME	for C. Parrish	DNeighbors:ds	SVarga	TNovak	CUTCHIN		
DATE	6/8/81	6/6/81	6/ /81	6/9/81	6/10/81		

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

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and 50-281

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The Unit 2 amendment revises the Technical Specifications to reflect that the Unit 1 change above applies to both Units 1 and 2. ~~The Unit 2 Technical Specification changes were issued in Amendment No. 59 on August 1, 1980.~~

The Unit 1 changes are the same as the changes made for Unit 2 in amendment No. 59 on August 1, 1980, and therefore the Safety Evaluation (SE) supporting the Unit 2 changes applies to Unit 1 also. The August 1, 1980 SE states this. Therefore, we conclude that these changes are acceptable.

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JUN 23 1981

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Executive Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

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In preparing these amendments, we discovered that Amendment Nos. 66 and 65 dated February 25, 1981 inadvertently deleted paragraph 3.F from the licenses of Units 1 and 2. Paragraph 3.F was already deleted for Unit 2 by Amendment 59 dated August 1, 1980. This current action now correctly deletes Paragraph 3.F for Unit 1.

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD		
SURNAME	[illegible]	DNeighbors:ds	SVarga	TNovak	[illegible]		
DATE	6/[illegible]/81	6/5/81	6/[illegible]/81	6/9/81	6/[illegible]/81		

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 71 to DPR-32
2. Amendment No. 71 to DPR-37
3. Notice of Issuance

cc: w/enclosures
See next page

Mr. J. H. Ferguson
Virginia Electric and Power Company

cc: Mr. Michael W. Maupin
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Donald J. Burke, Resident Inspector
Surry Power Station
U. S. Nuclear Regulatory Commission
Post Office Box 166
Route 1
Surry, Virginia 23883

Mr. Sherlock Holmes, Chairman
Board of Supervisors of Surry County
Surry County Courthouse, Virginia 23683

Commonwealth of Virginia
Council on the Environment
903 Ninth Street Office Building
Richmond, Virginia 23219

Attorney General
1101 East Broad Street
Richmond, Virginia 23219

Mr. James R. Wittine
Commonwealth of Virginia
State Corporation Commission
Post Office Box 1197
Richmond, Virginia 23209

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region III Office
ATTN: EIS COORDINATOR
Curtis Building - 6th Floor
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

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. 71
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 May 19, 1981, 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, Facility Operating License No. DPR-32 is hereby amended by deleting paragraph 3.F, by revising paragraph 3.B to read as follows, and by changing the Technical Specifications as indicated in the attachment to this license amendment.

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 71 , 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

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 23, 1981

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. 71
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 May 19, 1981, 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. 71 , 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

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 23, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages		Insert Pages
DPR-32	DPR-37	DPR-32 & DPR-37
3.3-1	3.3-1	3.3-1
3.3-2	3.3-2	3.3-2
3.3-3	3.3-3	3.3-3
3.3-4	3.3-4	3.3-4
3.3-5	3.3-5	3.3-5
3.3-6	3.3-6	3.3-6
3.3-7	3.3-7	3.3-7
3.3-8	3.3-8	3.3-8
3.3-9	3.3-9	3.3-9
3.4-1	3.4.2-1	3.4-1
3.4-2	3.4.2-2	3.4-2
3.4-3	3.4.2-3	3.4-3
3.4-4	3.4.2-4	3.4-4
3.4-5	3.4.2-5	3.4-5
------	-------	3.4-6
3.6-3	3.6-3	3.6-3
3.8-1	3.8.2-1	3.8-1
3.8-2	3.8.2-2	3.8-2
3.8-3	3.8.2-3	3.8-3
3.8-4	3.8.2-4	3.8-4
T.S. Figure 3.8-1	T.S. Figure 3.8.2-1	T.S. Figure 3.8-1
------	T.S. Figure 3.8.2-1 (cont)	T.S. Figure 3.8-1 (cont)
4.1-7	4.1-7	4.1-7
------	4.5-6	4.5-6
------	4.11-5	4.11-5

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated cooling water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless the following conditions are met:

1. The refueling water storage tank contains not less than 387,100 gal of borated water. The boron concentration shall be at least 2000 ppm and not greater than 2200 ppm.
2. Each accumulator system is pressurized to at least 600 psia and contains a minimum of 975 ft^3 and a maximum of 989 ft^3 of borated water with a boron concentration of at least 1950 ppm.
3. The boron injection tank and isolated portion of the inlet and outlet piping contains no less than 900 gallons of water with a boron concentration equivalent to at least 11.5% to 13% weight boric acid solution at a temperature of at least 145°F. Additionally, recirculation between a unit's Boron Injection Tank and the Boric Acid Tank(s) assigned to the unit shall be maintained.

4. Two channels of heat tracing shall be available for the flow paths.

5. Two charging pumps are operable.

6. Two low head safety injection pumps are operable.

7. All valves, piping, and interlocks associated with the above components which are required to operate under accident conditions are operable.

8. The Charging Pump Cooling Water Subsystem shall be operating as follows:

   a. Make-up water from the Component Cooling Water Subsystem shall be available.

   b. Two charging pump component cooling water pumps and two charging pump service water pumps shall be operable.

   c. Two charging pump intermediate seal coolers shall be operable.

9. During power operation the A.C. power shall be removed from the following motor operated valves with the valve in the open position:

Unit No. 1	Unit No. 2
MOV 1890C	MOV 2890C

10. During power operation the A.C. power shall be removed from the following motor operated valves with the valve in the closed position:

Unit No. 1	Unit No. 2
MOV 1869A	MOV 2869A
MOV 1869B	MOV 2869B
MOV 1890A	MOV 2890A
MOV 1890B	MOV 2890B

11. The accumulator discharge valves listed below in non-isolated loops shall be blocked open by de-energizing the valve motor operator when the reactor coolant system pressure is greater than 1000 psig.

Unit No. 1	Unit No. 2
MOV 1865A	MOV 2865A
MOV 1865B	MOV 2865B
MOV 1865C	MOV 2865C

12. Power operation with less than three loops in service is prohibited. The following loop isolation valves shall have AC power removed and be locked in open position during power operation.

Unit No. 1	Unit No. 2
MOV 1590	MOV 2590
MOV 1591	MOV 2591
MOV 1592	MOV 2592
MOV 1593	MOV 2593
MOV 1594	MOV 2594
MOV 1595	MOV 2595

13. The total system uncollected leakage from valves, flanges, and pumps located outside containment shall not exceed the limit shown in Table 4.11-1 as verified by inspection during system testing. Individual component leakage may exceed the design value given in Table 4.11-1 provided that the total allowable system uncollected leakage is not exceeded.

B. The requirements of Specification 3.3-A may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of Specification 3.3-A within the time period specified, the reactor shall initially be placed in the shutdown condition. If the requirements of Specification 3.3-A are not satisfied within an additional 48 hours the reactor shall be placed in cold shutdown condition.

1. One accumulator may be isolated for a period not to exceed 4 hours.
2. Two charging pumps per unit may be out service, provided immediate attention is directed to making repairs and one pump is restored to operable status within 24 hours.
3. One low head safety injection pump per unit may be out of service, provided immediate attention is directed to making repairs and the pump is restored to operable status within 24 hours. The other low head safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump and shall be tested once every eight (8) hours thereafter, until both pumps are in an operable status or the reactor is shutdown.
4. Any one valve in the Safety Injection System may be inoperable provided repairs are initiated immediately and are completed within 24 hours. Prior to initiating repairs, all automatic valves in the redundant system shall be tested to demonstrate operability.
5. One channel of heat tracing may be inoperable for a period not to exceed 24 hours, provided immediate attention is directed to making repairs.

6. One charging pump component cooling water pump or one charging pump service water pump may be out of service provided the pump is restored to operable status within 24 hours.

7. One charging pump intermediate seal cooler or other passive component may be out of service provided the system may still operate at 100 percent capacity and repairs are completed within 48 hours.

8. Power may be restored to any valve referenced in Specifications 3.3.A.9 and 3.3.A.10 for the purpose of valve testing or maintenance provided that no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours.

9. Power may be restored to any valve referenced in Specification 3.3.A.11 for the purpose of valve testing or maintenance provided that no more than one valve has power restored and provided that testing or maintenance is completed and power removed within 4 hours.

10. Recirculation between a unit's Boron Injection Tank and the Boric Acid Tank(s) assigned to the unit may be terminated for a period not to exceed two hours, provided all other parameters (temperatures, boron concentration, volume) of the Boron Injection Tank are within Specification 3.3.A.3 and immediate attention is directed to making repairs.

11. The total uncollected system leakage for valves, flanges, and pumps located outside containment can exceed the limit shown in Table 4.11-1 provided immediate attention is directed to making repairs and system leakage is returned to within limits within 7 days.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then make critical by withdrawing control rods and/or diluting boron in the coolant. With this mode of startup the Safety Injection System is required to be operable as specified. During low power physics tests there is a negligible amount of energy stored in the system; therefore an accident comparable in severity to the Design Basis Accident is not possible, and the full capacity of the Safety Injection System is not required.

The operable status of the various systems and components is to be demonstrated by periodic tests, detailed in TS Section 4.1. A large fraction of these tests are performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. A single component being inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component and, in some cases are to be retested at intervals during the repair period. In some cases, i.e. charging pumps, additional components are installed to allow a component to be inoperable without affecting system redundancy. For those cases which are not so designed, if it develops that (a) the inoperable component is not repaired within the specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. After 48 hours in the hot shutdown condition, if the malfunction(s) are not corrected the reactor will be placed in cold shutdown condition, following normal shutdown and cooldown procedures.

The Specification requires prompt action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. Furthermore, the specified repair times do not apply to regularly scheduled maintenance of the Safety Injection System, which is normally to be performed during refueling shutdowns. The limiting times for repair are based on: estimates of the time required to diagnose and correct various postulated malfunctions using safe and proper procedures, the availability of tools, materials and equipment; health physics requirements and the extent to which other systems provide functional redundancy to the system under repair.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat production decreases as follows after initiating hot shutdown.

Time After Shutdown	Decay Heat, % of Rated Power
1 min.	3.7
30 min.	1.6

Time After Shutdown	Decay Heat, % of Rated Power
1 hour	1.3
8 hours	0.75
48 hours	0.48

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is reduced by orders of magnitude below the requirements for handling a postulated loss-of-coolant accident occurring during power operation. Placing and maintaining the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, allows access to some of the Safety Injection System components in order to effect repairs, and minimizes the exposure to thermal cycling.

Failure to complete repairs within 48 hours of going to hot shutdown condition is considered indicative of unforeseen problems, i.e., possibly the need of major maintenance. In such a case the reactor is to be put into the cold shutdown condition.

The accumulators are able to accept leakage from the Reactor Coolant System without any effect on their availability. Allowable inleakage is based on the volume of water that can be added to the initial amount without exceeding the volume given in Specification 3.3.A.2. The maximum acceptable inleakage is 14 cubic feet per tank.

The accumulators (one for each loop) discharge into the cold leg of the reactor coolant piping when Reactor Coolant System pressure decreases below accumulator pressure, thus assuring rapid core cooling for large breaks. The line from each accumulator is provided with a motorized valve to isolate the accumulator during reactor start-up and shutdown to preclude the discharge of the contents of the accumulator when not required. These valves receive a signal to open when safety injection is initiated.

To assure that the accumulator valves satisfy the single failure criterion, they will be blocked open by de-energizing the valve motor operators when the reactor coolant pressure exceeds 1000 psig. The operating pressure of the Reactor Coolant System is 2235 psig and safety injection is initiated when this pressure drops to 600 psia. De-energizing the motor operator when the pressure exceeds 1000 psig allows sufficient time during normal startup operation to perform the actions required to de-energize the valve. This procedure will assure that there is an operable flow path from each accumulator to the Reactor Coolant System during power operation and that safety injection can be accomplished.

The removal of power from the valves listed in the specification will assure that the systems of which they are a part satisfy the single failure criterion.

Continuous recirculation between the Boron Injection Tank and the Boric Acid Tank(s) ensures that a unit's Boron Injection Tank is full of concentrated boric acid at all times.

3.4 SPRAY SYSTEMS

Applicability

Applies to the operational status of the Spray Systems.

Objective

To define those conditions of the Spray Systems necessary to assure safe unit operation.

Specification

A. A unit's Reactor Coolant System temperature or pressure shall not be made to exceed 350°F or 450 psig, respectively, or the reactor shall not be made critical unless the following Spray System conditions in the unit are met:

1. Two Containment Spray Subsystems, including containment spray pumps and motor drives, piping, and valves shall be operable.
2. Four Recirculation Spray Subsystems, including recirculation spray pumps, coolers, piping, and valves shall be operable.
3. The refueling water storage tank shall contain not less than 387,100 gal and not greater than 398,000 gal of borated water at a maximum temperature as shown in TS Fig. 3.8-1

   If this volume of water cannot be maintained by makeup, or the temperature maintained below that specified in TS Fig. 3.8-1, the reactor shall be shutdown until repairs can be made. The water shall be borated to a boron concentration not less than

2,000 ppm and not greater than 2,200 ppm which will assure that the reactor is in the refueling shutdown condition when all control rod assemblies are inserted.

4. The refueling water chemical addition tank shall contain not less than 4,200 gal of solution with a sodium hydroxide concentration of not less than 17 percent by weight and not greater than 18 percent by weight.

5. All valves, piping, and interlocks associated with the above components which are required to operate under accident conditions shall be operable.

6. The total uncollected system leakage from valves, flanges, and pumps located outside containment shall not exceed the limit shown in Table 4.5-1 as verified by inspection during system testing. Individual component leakage may exceed the design value given in Table 4.5-1 provided that the total allowed system uncollected leakage is not exceeded.

B. During power operation the requirements of Specification 3.4-A may be modified to allow the following components to be inoperable. If the components are not restored to meet the requirements of Specification 3.4-A within the time period specified below, the reactor shall be placed in the hot shutdown condition. If the requirements of Specification 3.4-A are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition using normal operating procedures.

1. One Containment Spray Subsystem may be out of service, provided immediate attention is directed to making repairs and the subsystem can be restored to operable status within 24 hours. The other Containment Spray Subsystem shall be tested as specified in Specification 4.5-A to demonstrate operability prior to initiating repair of the inoperable system.
2. One outside Recirculation Spray Subsystem may be out of service provided immediate attention is directed to making repairs and the subsystem can be restored to operable status within 24 hours. The other Recirculation Spray subsystem shall be tested as specified in Specification 4.5-A to demonstrate operability prior to initiating repair of the inoperable system.
3. One inside Recirculation Spray Subsystem may be out of service provided immediate attention is directed to making repairs and the subsystem can be restored to operable status within 72 hours. The other Recirculation Spray subsystems shall be tested as specified in Specification 4.5-A to demonstrate operability prior to initiating repair of the inoperable subsystems.
4. The total uncollected system leakage from valves, flanges, and pumps located outside containment can exceed the limit shown in Table 4.5-1 provided immediate attention is directed to making repairs and system leakage is returned to within limits within 7 days.

C. Should the refueling water storage tank temperature fail to be maintained at or below 45°F, the containment pressure and temperature shall be maintained in accordance with TS Fig. 3.8-1 to maintain the capability of the Spray System with the higher refueling water temperature. If the containment temperature and pressure cannot be maintained within the limits of TS Fig. 3.8-1, the reactor shall be placed in the cold shutdown condition.

Basis

The Spray Systems in each reactor unit consist of two separate parallel Containment Spray Subsystems, each of 100 percent capacity, and four separate parallel Recirculation Spray Subsystems, each of 50 percent capacity.

Each Containment Spray Subsystem draws water independently from the 398,000 gal. capacity refueling water storage tank. The water in the tank is cooled to 45°F or below by circulating the tank water through one of the two refueling water storage tank coolers through the use of one of the two refueling water recirculating pumps. The water temperature is maintained by two mechanical refrigerating units as required. In each Containment Spray Subsystem, the water flows from the tank through an electric motor driven containment spray pump and is sprayed into the containment atmosphere through two separate sets of spray nozzles. The capacity of the Spray Systems to depressurize the containment in the event of a Design Basis Accident is a function of the pressure and temperature of the containment atmosphere, the service water temperature, and the temperature in the refueling water storage tanks as discussed in Specification 3.8-B.

Each Recirculation Spray Subsystem draws water from the common containment pump. In each subsystem the water flows through a recirculation spray pump and recirculation spray cooler, and is sprayed into the containment atmosphere through a separate set of spray nozzles. Two of the recirculation spray pumps are located inside the containment and two outside the containment in the containment auxiliary structure.

With one Containment Spray Subsystem and two Recirculation Spray Subsystems operating together, the Spray Systems are capable of cooling and depressurizing the containment to subatmospheric pressure in less than 60 minutes following the Design Basis Accident. The Recirculation Spray Subsystems are capable of maintaining subatmospheric pressure in the containment indefinitely following the Design Basis Accident when used in conjunction with the Containment Vacuum System to remove any long term air in leakage.

In addition to supplying water to the Containment Spray System, the refueling water storage tank is also a source of water for safety injection following an accident. This water is borated to a concentration which assures reactor shutdown by approximately 10 percent Δk/k when all control rod assemblies are inserted and when the reactor is cooled down for refueling.

References

FSAR Section 4	Reactor Coolant System
FSAR Section 6.3.1	Containment Spray Subsystem
FSAR Section 6.3.1	Recirculation Spray Pumps and Coolers
FSAR Section 6.3.1	Refueling Water Chemical Addition Tank
FSAR Section 6.3.1	Refueling Water Storage Tank
FSAR Section 14.5.2	Design Basis Accident
FSAR Section 14.5.5	Containment Transient Analysis

450 psig, respectively, residual heat removal requirements are normally satisfied by steam bypass to the condenser. If the condenser is unavailable, steam can be released to the atmosphere through the safety valves, power operated relief valves, or the 4 inch decay heat release line.

The capability to supply feedwater to the generators is normally provided by the operation of the Condensate and Feedwater Systems. In the event of complete loss of electrical power to the station, residual heat removal would continue to be assured by the availability of either the steam driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps and the 110,000 gallon condensate storage tank.

A minimum of 92,000 gallons of water in the 110,000 gallon condensate tank is sufficient for 8 hours of residual heat removal following a reactor trip and loss of all off-site electrical power. If the protected condensate storage tank level is reduced to 60,000 gallons, the immediately available replenishment water in the 300,000 gallon condensate tank can be gravity-feed to the protected tank if required for residual heat removal. An alternate supply of feedwater to the auxiliary feedwater pump suction is also available from the Fire Protection System Main in the auxiliary feedwater pump cubicle.

The five main steam code safety valves associated with each steam generator have a total combined capacity of 3,725,575 pounds per hour at their individual set pressure; the total combined capacity of all fifteen main steam code safety valves is 11,176,725 pounds per hour. The ultimate power rating steam flow is 11,167,923 pounds per hour. The combined capacity of the safety valves required by Specification 3.6 always exceeds the total steam flow corresponding to the maximum steady-state power than can be obtained during one, two or three reactor

3.8 CONTAINMENT

Applicability

Applies to the integrity and operating pressure of the reactor containment.

Objective

To define the limiting operating status of the reactor containment for unit operation.

Specification

A. Containment Integrity and Operating Pressure

1. The containment integrity, as defined in TS Section 1.0, shall not be violated, except as specified in Specification 3.8.A.2, below, unless the reactor is in the cold shutdown condition.
2. The reactor containment shall not be purged while the reactor is operating, except as stated in Specification 3.8.A.3.
3. During the plant startup, the remote manual valve on the steam jet air ejector suction line may be open, if under administrative control, while containment vacuum is being established. The Reactor Coolant System temperature and pressure must not exceed 350°F and 450 psig, respectively, until the air partial pressure in the containment has been reduced to a value equal to, or below, that specified in TS Fig. 3.8-1.
4. The containment integrity shall not be violated when the reactor vessel head is unbolted unless a shutdown margin greater than 10 percent $\Delta k/k$ is maintained.

5. Positive reactivity changes shall not be made by rod drive motion or boron dilution unless the containment integrity is intact.

B. Internal Pressure

1. If the internal air partial pressure rises to a point 0.25 psi above the allowable value of the air partial pressure (TS Fig. 3.8-1), the reactor shall be brought to the hot shutdown condition.
2. If the leakage condition cannot be corrected without violating the containment integrity or if the internal partial pressure continues to rise, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures.
3. If the internal pressure falls below 8.25 psia the reactor shall be placed in the cold shutdown condition.
4. If the air partial pressure cannot be maintained greater than or equal to 9.0 psia, the reactor shall be brought to the hot shutdown condition.

Basis

The Reactor Coolant System temperature and pressure being below 350°F and 450 psig, respectively, ensures that no significant amount of flashing steam will be formed and hence that there would be no significant pressure build-up in the containment if there is a loss-of-coolant accident.

The shutdown margins are selected based on the type of activities that are being carried out. The 10 percent Δk/k shutdown margin during refueling precludes criticality under any circumstance, even though fuel and control rod assemblies are being moved.

The allowable value for the containment air partial pressure is presented in TS Fig. 3.8-1 for service water temperatures from 25 to 90°F. The allowable value varies as shown in TS Fig. 3.8-1 for a given containment average temperature. The RWST water shall have a maximum temperature of 45°F.

The horizontal limit lines in TS Fig. 3.8-1 are based on LOCA peak calculated pressure criteria, and the sloped line is based on LOCA subatmospheric peak pressure criteria.

The curve shall be interpreted as follows:

> The horizontal limit line designates the allowable air partial pressure value for the given average containment temperature. The horizontal limit line applies for service water temperatures from 25°F to the sloped line intersection value (maximum service water temperature).

From TS Fig. 3.8-1, if the containment average temperature is 112°F and the service water temperature is less than or equal to 83°F, the allowable air partial pressure value shall be less than or equal to 9.65 psia. If the average containment temperature is 116°F and the service water temperature is less than or equal to 88°F, the allowable air partial pressure value shall be less than or equal to 9.35 psia. These horizontal limit lines are a result of the higher allowable initial containment average temperatures and the analysis of the pump suction break.

If the containment air partial pressure rises to a point 0.25 psi above the allowable value, the reactor shall be brought to the hot shutdown condition. If a LOCA occurs at the time the containment air partial pressure is 0.25 psi above the allowable value, the maximum containment pressure will be less than 45 psig, the containment will depressurize in less than 1 hour, and the maximum subatmospheric peak pressure will be less than 0.0 psig.

If the containment air partial pressure cannot be maintained greater than or equal to 9.0 psia, the reactor shall be brought to the hot shutdown condition. The shell and dome plate liner of the containment are capable of withstanding an internal pressure as low as 3 psia, and the bottom mat liner is capable of withstanding an internal pressure as low as 8 psia.

References

FSAR Section 4.3.2	Reactor Coolant Pump
FSAR Section 5.2	Containment Isolation
FSAR Section 5.2.1	Design Bases
FSAR Section 5.5.2	Isolation Design

ALLOWABLE AIR PARTIAL PRESSURE
SURRY POWER STATION

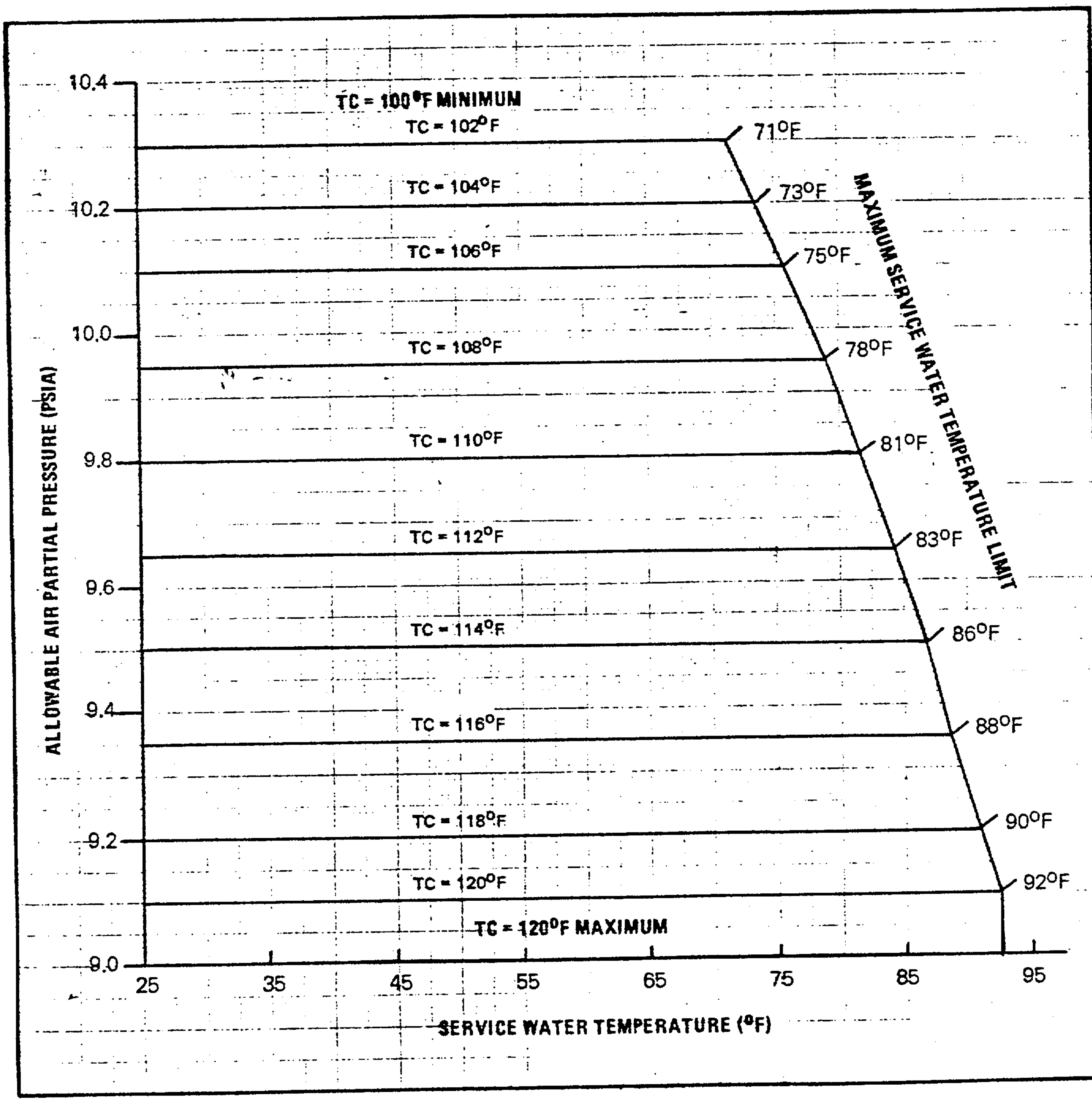

FIGURE 3.8-1 (Continued)

FIGURE NOTATION

TC - Containment average temperature.

FIGURE NOTES

1. Allowable operating air partial pressure in the containment as a function of service water temperature.
2. Refueling Water Storage Tank temperature ≦ 45°F.
3. Horizontal lines designate allowable air partial pressure setpoint per given containment average temperature.
4. Each containment temperature line is a maximum for the given air partial pressure.
5. Hot shutdown is required for containment air partial pressure increase greater than 0.25 psi above the allowable value or less than 9.0 psia.
6. Cold shutdown is required for containment air partial pressure less than 8.25 psia.

TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S (1,2)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heal Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	*D	R	N.A.	
15.	Refueling Water Storage Tank Level	S	R	M	
16.	Boron Injection Tank Level	W	N.A.	N.A.	
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	Reactor Containment Pressure-CLS	*D	R	M (1)	1) Isolation Valve signal and spray signal
19.	Process and Area Radiation Monitoring System	*D	R	M	
20.	Boric Acid Control	N.A.	R	N.A.	
21.	Containment Pump Level	N.A.	R	N.A.	
22.	Accumulator Level and Pressure	S	R	N.A.	
23.	Containment Pressure-Vacuum Pump System	S	R	N.A.	
24.	Steam Line Pressure	S	R	M	

TABLE 4.5-1
RECIRCULATION SUBSYSTEM LEAKAGE*

Item	No. of Units	Type of Leakage Control and Unit Leakage Rate	Design Uncollected Leakage cc per hr**	Leakage to Vent and Drain System, cc per hr
Recirculation spray pumps	2	No leak of spray water due to tandem seal arrangement	0	0
Flanges:		40 drops per min per flange		
a. pump	4		480	0
b. Valves - bonnet to body (larger than 2 in.)	4		460	0
Valves - Stem leakoffs	4	Backseated, double packing with leakoff - 4 cc per hr per in. stem diameter	0	16
Miscellaneous small valves	2	Flanges body, packed stem - 4 drop per min	24	0
		Total	964	16

*Based on two subsystems in operation under DBA conditions.

Total Allowed System Uncollected Leakage is 964.cc/hr.

**Individual component uncollected leakage may exceed the design value provided that the total allowable system uncollected leakage is not exceeded.

TABLE 4.11-1

EXTERNAL RECIRCULATION LOOP LEAKAGE (Safety Injection System Only)

Item	No. of Units	Type of Leakage Control and Unit Leakage Rate	Design Leakage to Atmosphere cc per hr**	Design Leakage to Waste Disposal Tank, cc per hr
Low Head Safety Injection Pumps	2	Mechanical Seal with leakoff - 4 drop per min	0	24
Safety Injection Charging	3	Mechanical Seal with leakoff - 4 drop per min	0	36
Flanges:				
a. Pump	10	Gasket - adjusted to zero leakage following any test - 40 drop per min, per flange	1,200	0
b. Valves Bonnet to Body (larger than 2 in.)	54		2,240	0
Valves - Stem Leakoffs	27	Backseated, double packing with leakoff - 4 cc per hr per in stem diameter	0	108
Misc. Valves	33	Flanges body packed stems - 4 drop per min	396	0
		Totals	3,836	168

Total Allowed System Uncollected Leakage is 3,836 cc/hr

**Individual component uncollected leakage may exceed the design value provided that the total allowable system uncollected leakage is not exceeded.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 71 to Facility Operating License No. DPR-32 and Amendment No. 71 to Facility Operating License No. DPR-37 issued to Virginia Electric and Power Company (the licensee), which revised Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2, respectively, (the facilities), located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The Unit 1 amendment revises the Technical Specifications to reflect changes as a result of modifications made to alleviate Net Positive Suction Head (NPSH) problems with the Low Head Safety Injection and Recirculation Spray Pumps and modifications made to the containment spray system. Changes have been made to service water temperature, containment temperature, containment air partial pressure, refueling water storage tank volume and outside recirculation pump flow rate. These limits have been transferred to the Technical Specifications from the license.

The Unit 2 amendment revises the Technical Specifications to reflect that the Unit 1 change above applies to both Units 1 and 2. The Unit 2 Technical Specification changes were issued in Amendment No. 59 on August 1, 1980.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since these amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated May 19, 1981, (2) Amendment Nos. 71 and 71 to License Nos. DPR-32 and DPR-37, (3) the Commission's letter to the licensee dated June 23, 1981, and (4) the Commission's Safety Evaluation dated August 1, 1980. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia 23185. A copy of items (2), (3), and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 23 day of June, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing