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Docket No. 50-281

AUGUST 1 1980

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

Dear Mr. Ferguson:

The Commission has issued the enclosed Amendment No. 59 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated June 30, 1980.

The 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.

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:
S. A. Varga

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

Enclosures:
1. Amendment No. 59 to DPR-37
2. Safety Evaluation
3. Notice of Issuance

cc w/encl:
See next page

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#16 OFFICE	DL:ORB#1	DL:ORB#1	DL:OR	OELD	DL:ORB#1	
retype 7/25 SURNAME	JDNeighbors	CParrish	TNovak	MCutchin	SVarga	
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Docket No. 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:

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

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. to DPR-37
2. Safety Evaluation
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cc: w/enclosures
See next page

No Legal Objection per telecon 7/31/80

JDN 7/21/80

#16

OFFICE	DL:ORB1	DL:ORB1	~~DL:AD:OR~~	OELD	~~DL:ORB1~~
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 1, 1980

Docket No. 50-281

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

Dear Mr. Ferguson:

The Commission has issued the enclosed Amendment No. 59 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated June 30, 1980.

The 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.

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

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

Enclosures:
1. Amendment No.59 to DPR-37
2. Safety Evaluation
3. Notice of Issuance

cc w/encl:
See next page

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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
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State Corporation Commission
Post Office Box 1197
Richmond, Virginia 23209

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
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Arlington, Virginia 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. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 59
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 June 30, 1980, 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 deleting paragraph 3.F of the license and by changes to the Technical Specifications as indicated in the attachment to the license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 59 , 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: August 1, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 59

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and certain vertical lines indicating the area of change.

Remove

3.3-1
3.3-2
3.3-3
3.3-4
3.3-5
3.3-6
3.3-7
3.3-8
3.3-9

3.4-1
3.4-2
3.4-3
3.4-4
3.4-5

3.6-3

3.8-1
3.8-2
3.8-3
3.8-4
TS Figure 3.8-1
4.1-7

Insert

3.3-1
3.3-2
3.3-3
3.3-4
3.3-5
3.3-6
3.3-7
3.3-8
3.3-9

3.4.2-1
3.4.2-2
3.4.2-3
3.4.2-4
3.4.2-5
3.6-3

3.8.2-1
3.8.2-2
3.8.2-3
3.8.2-4
TS Figure 3.8.2-1
TS Figure 3.8.2-1 (Continued)
4.1-7
4.5-6
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 385,200 gal (Unit 1) or 387,100 gal (Unit 2) of borated water. For Unit 1 only, the boron concentration shall be at least 2000 ppm. For Unit 2 only, 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 1980C	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. The leakage limits are for each unit. The leakage limits are for each unit.

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 hot shutdown condition. If the requirements of Specification 3.3-A are not satisfied within an additional 48 hours the reactor shall be placed in the 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 of 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 pumps 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 3.3.A.9 and 3.3.A.10 for the purpose of valve testing or maintenance providing 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 3.3.A.11 for the purpose of valve testing or maintenance providing 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 made 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 is 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 the 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 psig. 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.

3.4.2 SPRAY SYSTEMS (UNIT 2)

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 Fig. 3.8.2-1.

   If this volume of water cannot be maintained by makeup, or the temperature maintained below that specified in TS Fig. 3.8.2-1, 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 2200 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 3.4.2-A may be modified to allow the following components to be inoperable. If the components are not restored to meet the requirements of Specification 3.4.2-A within the time period specified below, the reactor shall be placed in the hot shutdown condition. If the requirements of Specification 3.4.2-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 subsystems 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 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. 3.8.2-1 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. 3.8.2-1 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 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 3.8.2-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

3.8.2 CONTAINMENT (UNIT 2)

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 A2, 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 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 Figure 3.8.2-1.
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 maximum allowable set point value of the air partial pressure (TS Figure 3.8.2-1), 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. The minimum allowable set point for the air partial pressure is 9.1 psia. 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 maximum allowable set point for the containment air partial pressure is presented in Figure 3.8.2-1 for service water temperature from 25 to 90°F. The allowable set point varies as shown in Figure 3.8.2-1 for a given containment average temperature. The RWST water shall have a maximum temperature of 45°F.

The horizontal limit lines in Figure 3.8.2-1 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 maximum air partial pressure set point 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 Figure 3.8.2-1, if the containment average temperature is 112°F and the service water temperature is less than or equal to 83°F, the air partial pressure set point 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 air partial pressure set point 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 maximum set point 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 set point 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.

The minimum allowable set point for the containment air partial pressure is 9.1 psia. 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 o 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

MAXIMUM ALLOWABLE AIR PARTIAL PRESSURE

SURRY POWER STATION - UNIT NO. 2

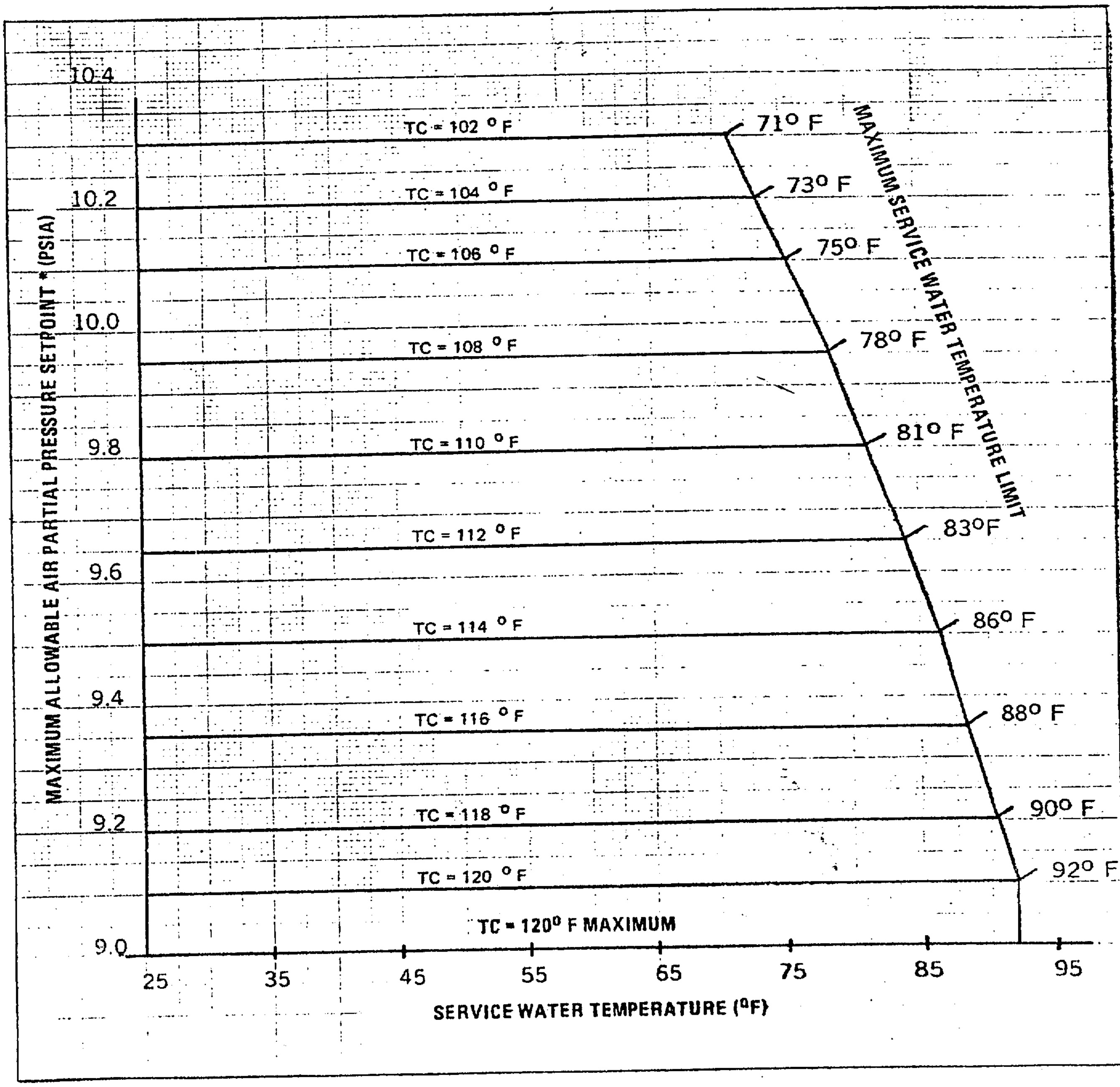

FIGURE 3.8.2-1 (Continued)

FIGURE NOTATION

* - Setpoint value in containment vacuum system instrumentation.

TC - Containment average temperature.

FIGURE NOTES

1. Maximum allowable operating air partial pressure in the containment as a function of service water temperature.
2. Refueling Water Storage Tank temperature $\leq 45^{o}F$.
3. Horizontal lines designate maximum 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 setpoint increase greater than 0.25 psi or less than 9.0 psia.
6. Cold shutdown is required for containment air partial pressure less than 8.25 psia.

	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 our of service 2) With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	*D	R	N.A.	
15A.	Unit 1 Refueling Water Storage Tank Level	W	R	N.A.	
15B.	Unit 2 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 Systems	*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)

Items	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 drops 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
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 59

FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NO. 2

DOCKET NO. 50-281

Introduction

By letter dated June 30, 1980, the Virginia Electric and Power Company (the licensee) requested an amendment to the Surry Power Station, Unit No. 2, license which would change the Technical Specifications. These changes were required because of changes to the recirculation spray pumps, low head safety injection pumps and the containment spray system.

The licensee's letter of June 30, 1980 provides a list of correspondence related to this evaluation.

Background

NPSH and Containment Pressure and Temperature Analyses

During the course of the operating license review of the North Anna Station, the licensee reevaluated the net positive suction head (NPSH) available to the recirculation spray (RS) and low head safety injection (LHSI) pumps based on a more conservative containment analysis. NPSH is the head, or potential energy, available or required to force a given flow into the impeller of a pump. NPSH is affected by containment pressure, sump water vapor pressure, depth of sump water and suction piping resistance to flow. The revised analysis incorporated analytical techniques and assumptions that were selected to minimize the containment pressure and maximize the containment sump water temperature, thereby minimizing the calculated NPSH available to the pumps; the other factors, namely, depth of sump water and suction piping resistance to flow, have a lesser affect on the revised analysis. As a result of the analysis,certain design modifications were found to be necessary to assure the adequacy of the available NPSH for both the RS and LHSI pumps.

The Surry Station, Units 1 and 2 are operating plants with a design similar to that of North Anna. It was determined that in the event of a major loss-of-coolant accident, the vapor pressure of the water in the Surry containment sump which is the source of water for the RS and LHSI pumps during the recirculation phase is higher than the original analyses had indicated. This situation can result in inadequate NPSH for the RS and LHSI pumps at specific times during the recirculation phase of long term core cooling and containment cooling.

By a letter dated August 24, 1977, the licensee proposed interim modifications of the RS and LHSI systems and requested that the Surry Power Station be permitted to operate with the proposed interim modifications until such time as permanent modifications are designed and installed. Based on our review of the information provided by the licensee, we found that the above proposed modifications were acceptable on a interim basis, and by Order dated August 24, 1977, we concluded that until permanent modifications are implemented, operation would not pose an undue threat to the health and safety of the public.

By a letter dated November 22, 1977, and June 30, 1980, the licensee submitted a report, which present: (1) proposed permanent modifications of the RS and LHSI systems; and (2) the containment pressure and temperature response analyses and associated NPSH available to the RS and LHSI pumps.

Containment Spray Modifications

In 1976, using the meteorological data from the Surry Power Station Units 3 and 4 (Surry 3/4) docket for the period March 3, 1974, to March 2, 1975, new accident relative concentration X/Q values were calculated by the staff for Surry 1/2. These X/Q values were higher than those used in the Surry 1/2 Safety Evaluation (SE) dated February 23, 1972. This prompted a request dated July 9, 1976, and February 1, 1977, to the licensee to supply the staff with additional information concerning the Surry 1/2 spray system and containment so that we could evaluate the containment spray system. The licensee responded to this request by letters dated August 31, 1976, and May 9, 1977, and June 30, 1980.

Modifications made to Surry Unit 2 are:

A. NPSH Modifications

1. Inside Recirculation Spray System

   a. Remove and plug all type 1HH30100 nozzles in the spray headers.
   b. Install and 2-1/2 in. bleed line from the discharge of the Recirculation Spray heat exchangers to the suction of the IRS pumps. Design flow is 350 gpm.

2. Outside Recirculation Spray System

   a. Remove and plug all type 1HH30100 nozzles in the containment recirculation spray headers.
   b. Install a restriction orifice on the ORS pump discharge to limit system flow to 3000 gpm.
   c. Install a 2-1/2 in. bleed line from each Containment Spray System supply header to the suction of the ORS pump in the containment sump. Design flow is 300 gpm.

3. Low Head Safety Injection System

   a. Install cavitating venturis in each of the cold leg injection lines to limit LHSI pump flow to 3250 gpm during the recirculation mode of operation.

4. Refueling Water Storage Tank (RWST)

   a. In conjunction with the RWST modifications for the Containment Spray (CS) Modification, elbows were installed inside the RWST on the CS pump suction lines.

B. Containment Spray System Modifications

1. Containment Spray Headers

   a. Install new containment spray header outside the crane wall.
   b. Replace nozzles in existing headers.

2. Caustic Addition Modifications

   a. Resize and reroute Chemical Addition Tank (CAT) outlet line directly to CS pump suction.

3. RWST Modifications

   a. Removal of mixing weir inside RWST
   b. Installation of elbows on CS pump suction lines inside RWST
   c. Upgrade of level instrumentation to provide input to control circuitry for automatic switchover of the LHSI system suction from the RWST to the containment sump.

The basis for implementing the above modifications was to (1) ensure adequate iodine removal for the most restrictive LOCA for all Engineered Safety Feature pump combinations (2) provide adequate spray to ensure containment depressurization for all pump combinations and (3) ensure adequate NPSH available for all LOCA transients. This has been accomplished by modifications to (1) provide increased caustic spray coverage, (2) reduce the delay time in caustic solution reaching the spray nozzles, (3) add caustic solution at a rate that will assure spray pH and sump pH is within bounds of the licensing requirements for all containment depressurization transients, (4) achieve maximum spray thermal effectiveness for the Containment and Recirculation Spray (RS) Systems, (5) reduce NPSH required for the LHSI and RS Systems by restricting maximum flow conditions, and (6) increase NPSH available for the RS Systems by providing subcooled water to pump suctions.

The above modification will be made to Surry Unit 1 during the outage for the steam generator repair and this evaluation will also apply to it when the same modifications are made.

Evaluation

NPSH AND CONTAINMENT ANALYSIS

The calculated pressure in the containment and temperature of the water that accummulates in the containment sumps are important parameters, in regard to available NPSH, in determining the RS and LHSI pump operability following a LOCA. These terms, in combination with the pump static head and associated line friction losses, establish the available NPSH during the transient.

The required NPSH may be reduced by a reduction in the pump flow rate. Alternately, the NPSH available at a given flow rate may be increased by the injection of cold water into the pump suction. The injection of cold water lowers the water temperature at the pump suction and, therefore, lowers the vapor pressure of the water entering the pump. The licensee proposed to utilize both of the above methods to resolve this problem.

Recirculation Spray Pumps

In order to assure an adequate amount of NPSH for the RS pumps, the licensee proposed:

(1) Diverting a portion (300 gpm) of the cold quench spray (QS) water from each of the QS headers to each of the outside RS pump suction piping; and

(2) Routing a bleed flow (350 gpm) from the discharge of the RS cooler back to the suction of the respective inside RS pump.

The cold QS water and the cool bleed flow injection will lower the water temperature at the pump suction and, thereby lower the vapor pressure of the water entering the pump.

A 2-1/2 inch line from each QS header inside the containment will be routed to the suction of each of the outside RS pumps on the same safety train as the QS pumps supplying the water. Also, a 2-1/3 inch line from the discharge of the RS cooler will be routed back to the suction of the respective inside RS pump. No active components will be used. This proposed modification will allow the RS pumps to perform with adequate NPSH and required RS flow rate.

Low Head Safety Injection Pumps

The change in the low safety injection flow was needed in order to meet the NPSH requirements of the LHSI pumps. The flow was limited by means of a venturi flow restrictor. This change resulted in a slightly lower safety injection flow. However, the licensee has demonstrated that this flow is still higher than the value assumed in the LOCA analysis and was evaluated in our Safety Evaluation on ECCS performance dated May 16, 1980.

CONTAINMENT ANALYSIS FOR THE EVALUATION OF AVAILABLE NPSH

The new containment response analysis submitted by the licensee to determine the containment pressure and sump water temperature response was based on the following.

The analytical technique used to determine the distribution of mass and energy in the liquid and vapor regions of the containment following a LOCA can influence the containment pressure/temperature response. The pressure flash method and temperature flash method are the two currently used techniques. For the NPSH analysis, the licensee used the pressure flash method which assumes that liquid being expelled from the break flashes at the saturation temperature corresponding to the containment total pressure. This maximizes the temperature of the water entering the sump, and is, therefore, conservative. Previously, the containment analytical model for NPSH analysis assumed that the liquid flashes at the dew point temperature of the containment atmosphere (temperature flash method). The temperature flash method is typically used for peak containment pressure calculations.

The pipe break effluent was assumed to be uniformly mixed with the ECCS injection water spilling from the break. This is an important consideration for postulated cold leg breaks and essentially increases the energy transferred to the sump. This assumption does not affect NPSH calculations for postulated hot leg breask since the break effluent is already uniformly mixed. Previously, for the NPSH analysis of postulating cold leg breaks, ECCS water was assumed to spill directly to the sump without mixing, which resulted in lower calculated sump water temperatures.

The licensee conducted a number of sensitivity studies to identify the other assumptions that should be used to minimize the calculated available NPSH. We have reviewed the results of these sensitivity studies and conclude that the following conservative assumptions will minimize the calculated available NPSH:

(1) A spray thermal effectiveness of 100% was assumed.

(2) A low initial containment pressure and high initial containment temperature were assumed.

Sensitivity studies were also done to identify the single failure, break size and pipe break location that will give the lowest calculated available NPSH for the RS and LHSI pumps. The results of these studies indicated that for the RS pumps, a postulated hog leg double-ended rupture will result in the lowest available NPSH, and for the LHSI pumps a postulated pump suction double-ended pipe rupture will result in the lowest available NPSH. The available NPSH for the inside recirculation pumps was calculated to be 15.0 feet, the available NPSH for the outside recirculation pumps was calculated to be 11.9 feet and the available NPSH for the LHSI pumps was calculated to be 17.2 feet. The minimum NPSH required are 8.4 feet for the outside RS pumps; 10.1 feet for the inside RS pumps; and 15.2 feet for the LHSI pumps.

We have performed confirmatory analyses for the pipe break locations that the licensee has identified as giving the lowest available NPSH for the pumps. For our confirmatory analyses, we used CONTEMPT (MOD26) computer code. The code has been modified to permit the analyses to be based on the pressure flash method. The results of our analysis; i.e., the containment pressure and sump water temperature versus time, are in good agreement with the licensee's results. We, therefore, conclude that the licensee's NPSH analysis is acceptable.

Effects on Containment Depressurization

In view of the system modifications that were found necessary to satisfy the NPSH requirements of the RS and LHSI pumps, the licensee also performed a sensitivity study to determine the impact on the depressurization time used in performing the analysis of the radiological consequences following a postulated loss-of-coolant accident. The results indicate that the containment will be depressurized to below a atmospheric pressure within an hour following a LOCA.

The limiting case for containment depressurization is a pump suction double-ended rupture with minimum engineered safety feature operation. A depressurization time of 45.3 minutes was calculated, which is less than the one hour used in performing the analysis of the radiological consequences following a LOCA. We have reviewed the input parameters used by the licensee to perform the depressurization analysis and concluded that the analysis would result in a reasonably conservative calculation of the containment depressurization time. Therefore, we conclude that the licensee's containment depressurization analysis is acceptable.

CONTAINMENT SPRAY MODIFICATIONS

Based on our requests for information on the containment spray systems and on our discussions on this system, the licensee modified several containment spray components. These modifications will provide additional assurance that the potential consequences of the postulated LOCA are below the guidelines of 10 CFR Part 100. This included adding restricting flow orifices to the lines that carry NaOh solution from the chemical addition tank (CAT) to each of the two containment spray (CS) trains. By controlling the volume flow rate of the gravity-fed caustic solution, the pH of containment spray and the recirculation water in the sump can be kept within acceptable limits. A drawdown test of the new system was used to test the components and determine the head loss coefficients for the CAT and refueling water storage tank (RWST) lines. The results of this test were used by the licensee to predict CS pH under a variety of limiting operating conditions. The licensee's proposed Technical Specification changes ensure that CS pH will be within current Standard Review Plan (SRP) 6.5.2 limits.

The licensee's drawdown test and application of hydraulic and chemical models predict that the containment spray (but not the recirculation spray) will have a pH within 8.5 to 11, as recommended in SRP 6.5.2. To keep the pH within these limits, under a variety of operating conditions, Technical Specifications were proposed to require narrower limits on volumes and concentrations in the RWST and CAT. These constraints, and the demonstration of the predictable nature of the gravity feed flow from the CAT, provide reasonable assurance that the pH can be kept within those limits.

However, for the case of the CAT drawing down at the maximum rate, the CAT will be effectively empty at about 42 minutes after the initiation of spray, and from then on the containment spray will be less effective in removing iodine. Since the Technical Specifications require the capability for reduction of containment pressure to subatmospheric within 60 minutes following a LOCA, there could be a period between 42 and 60 minutes when the containment would be at a positive pressure and there would not be a highly effective containment spray. Our analysis (see Tables 1, 2 and 3) indicates that the 0-2 hr. exclusion area boundary (EAB) dose for a LOCA and containment ECCS leakage exceeds 10 CFR Part 100 limits when no credit is taken for recirculation spray. The recirculation spray starts after about five minutes, with an initial pH of 7.0, which slowly rises as containment spray water mixes with ECCS water. However, after 42 minutes, the sump water is predicted to have only reached a pH of 8.0 (lower than the pH 8.5 recommended in SRP 6.5.2). Elemental iodine removal coefficients were conservatively estimated to be 10 hr^{-1} for pH>8.0, and 5 hr^{-1} for 7.0<pH<8.0, for the volume covered by recirculation spray. Using these removal coefficients, our analysis (see Tables 1, 2 and 3) indicated that therewwould be adequate iodine removal to keep the doses resulting from the worst case design basis accident within 10 CFR Part 100 guidelines. The relative concentration X/Q values used in the analyses were calculated by us for Surry 1 and 2 in 1979. These values were higher than the values in the SE dated February 23, 1972, but lower than the values calculated by us in 1976.

In order to provide additional assurance that these evaluations are valid, the licensee will submit Technical Specifications for engineered safety system ventilation filters by October 1, 1980. Based on the above, we conclude that the potential radiological dose consequences of the postulated LOCA at Surry 2 are below the 10 CFR Part 100 guidelines and are therefore acceptable.

Technical Specifications

We have evaluated the proposed Technical Specifications and conclude that they adequately incorporate the requirements evaluated herein, and when the modifications are made on Unit 1 as on Unit 2, this evaluation will also apply.

Environmental Consideration

We have determined that the amendment does 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 amendment involves 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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: August 1, 1980

TABLE 1

ASSUMPTIONS USED IN THE LOCA DOSE REEVALUATION

Power Level	2490 MWt
Fraction of Core Inventory Available for Leakage	
Iodines	25 Percent
Noble Gases	100 Percent
Initial Iodine Composition in Containment	
Elemental	91 Percent
Organic	4 Percent
Particulate	5 Percent
Containment Free Volume	1.753×10^6 ft^3
Containment Volume:	
Sprayed Fraction, Containment (quench) spray	0.72
Sprayed Fraction, Recirculation Spray	0.14
Unsprayed Fraction	0.14
Containment Mixing Rate Between Sprayed and Unsprayed Volumes	2.0 Unsprayed Volumes per hour
Spray Removal Coefficients for Containment Spray (Quench Spray)	
Elemental Iodine	10 per hour
Particulate Iodine	0.45 per hour
Organic Iodine	0
Spray Initiation Delay Time	0 seconds
Duration	42 minutes

TABLE 1 (cont'd)

ASSUMPTIONS USED IN THE LOCA DOSE REEVALUATION (cont'd)

Spray Removal Coefficients for recirculation Spray	
Elemental Iodine, 5-42 minutes ($7.0 < pH < 8.0$)	5 per hour
42-60 minutes ($8.0 < pH < 8.5$)	10 per hour
Particulate Iodine	.45 per hour
Organic Iodine	0 per hour
Spray Initiation Delay Time	5 minutes
Duration	5 minutes to indef.
0.5% Sector Probability Direction-Dependent X/Q Values	
Exclusion Area Boundary (NE @ 520m)	
0 - 1 Hour	1.6×10^{-3} sec/m^3
Low Population Zone (NE @ 4828m)	
0 - 1 Hour	1.6×10^{-4} sec/m^3
Containment Leak Rate	
0 - 60 minutes	0.1 Percent per day
> 60 minutes	0.0 Percent per day

TABLE 2

ASSUMPTIONS USED IN THE ECCS LEAKAGE ANALYSIS

Power Level	2490 MWt
Fraction of Core Iodine Inventory in Containment Sump	50 Percent
Volume of Containment Sump Water	56117.0 ft^3
Volume of Sump Water not recirculated (10%)	5611.7 ft^3
Iodine Decontamination Factor	10
Filter Efficiency for Iodine	90 Percent

ECCS Leak Rates Outside Containment

Time	Leak Rate*
0 - 5 minutes	0
5 - 20 minutes	5.7×10^{-4} ft^3/min
20 min - 30 days	2.825×10^{-3} ft^3/min

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* Twice the proposed technical specification leak rates were used for calculations.

TABLE 3

RESULTS OF LOCA DOSE ANALYSIS AT SURRY 2

Using 0.5% Sector Probability X/Q Values

I. Allowing Surry 2 Credit for Iodine Removal By Recirculation Sprays

	Containment Leakage		ECCS Leakage	
Exclusion Area Boundary	Thyroid Dose (Rem)	Whole Body Dose (Rem)	Thyroid Dose (Rem)	Whole Body Dose (Rem)
0 - 2 Hour	248	5	5	1
Low Population Zone				
0 - 30 Days	25	0.5	3	1

II. Allowing Surry 2 No Credit for Recirculation Spray Iodine Removal

	Containment Leakege		ECCS Leakage*	
Exclustion Area Boundary	Thyroid Dose (Rem)	Whole Body Dose (Rem)	Thyroid Dose (Rem)	Whole Body Dose (Rem)
0 - 2 Hours	307	6	5	1
Low Population Zone				
0 - 30 Days	31	0.6	3	1

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 59 to Facility Operating License No. DPR-37 issued to Virginia Electric and Power Company, which revised Technical Specifications for operation of the Surry Power Station, Unit No. 2 (the facility) located in Surry County, Virginia. The amendment is effective as of the date of issuance.

The 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 application for the amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does 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) and environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated June 30, 1980, (2) Amendment No. 59 to License No. DPR-37, and (3) the Commission's related Safety Evaluation. 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 the Swem Library, College of William and Mary, Williamsburg, Virginia. A copy of items (2) and (3) 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 1st day of August, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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