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Docket Nos. 50-280
and 50-281

Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 95 to Facility Operating License No. DPR-32 and Amendment No. 94 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 September 13, 1983, as supplemented October 6, November 30, and December 19, 1983, and January 18 and 25, 1984.

These amendments revise the Technical Specifications to change the minimum Boron Injection Tank boron concentration from 11.5% to 0% and to change the Boric Acid System boron concentration from 11.5% to 7%.

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

Sincerely,

/s/ Joseph D. Neighbors

Joseph D. Neighbors, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 95 to DPR-32
2. Amendment No. 94 to DPR-37
3. Safety Evaluation

cc w/enclosures:
See next page

note met-EP 2-24-8

ORB #1	ORB #1	ORB #1	OELD	AD:OL
CParrish/jm	DNeighbors	S Varga	JOHNSON	G Lainas
2/13/84	2/13/84	2/23/84	2/14/84	2/24/84

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PDR ADOCK 05000280
P PDR

Mr. W. L. Stewart
Virginia Electric and Power Company

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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. 95
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 September 13, 1983, as supplemented October 6, November 30, and December 19, 1983, and January 18 and 25, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 95 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented no later than March 9, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: FEB 24 1984

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. 94
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 September 13, 1983, as supplemented October 6 November 30, and December 19, 1983, and January 18 and 25, 1984, 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 94 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented no later than March 30, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: FEB 24 1984

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.2-1	3.2-1
3.2-2	3.2-2
3.2-4	3.2-4
3.2-5	3.2-5
3.3-1	3.3-1
3.3-2	3.3-2
3.3-3	3.3-3
3.3-5	3.3-5
3.3-9	3.3-9
4.1-7	4.1-7
4.1-8	4.1-8
4.1-9	4.1-9
4.1-10	4.1-10

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

A. When fuel is in a reactor there shall be at least one flow path to the core for boric acid injection. The minimum capability for boric acid injection shall be equivalent to that supplied from the refueling water storage tank.

B. For one unit operation the reactor shall not be critical unless the following Chemical and Volume Control System conditions are met:

1. Two charging pumps shall be operable.
2. Two boric acid transfer pumps shall be operable.
3. The boric acid tanks (tank associated with the unit plus the common tank) together shall contain a minimum of 6000 gallons of at least 7.0% (but not greater than 8.5%) by weight boric acid solution at a temperature of at least 112°F.

4. System piping and valves shall be operable to the extent of establishing two flow paths to the core; one flow path from the boric acid tanks to the charging pumps and a flow path from the refueling water storage tank to the charging pumps.
5. Two channels of heat tracing shall be operable for the flow paths requiring heat tracing.

C. For two unit operation the reactor shall not be critical unless the following Chemical and Volume Control System conditions are met:

1. Two charging pumps shall be operable per unit.
2. Three boric acid transfer pumps shall be operable.
3. When the common tank is in service, it shall be assigned to only one unit at a time. For that unit which has usage of the common tank, the boric acid tanks (unit's tank plus common tank) together shall contain a minimum of 6000 gallons of at least 7.0% (but not greater than 8.5%) by weight boric acid solution at a temperature of at least 112°F.

   For that unit which does not have usage of the common tank, the unit's own tank shall contain a minimum of 6000 gallons of at least 7.0% (but not greater than 8.5%) by weight boric acid solution at a temperature of at least 112°F.

   When the common tank is assigned to one unit, valves shall be positioned to establish a flow path to that unit and prevent flow to the other unit.

The Chemical and Volume Control System provides control of the Reactor Coolant System Boron inventory. This is normally accomplished by using boric acid transfer pumps which discharge to the suction of each unit's charging pumps. The Chemical and Volume Control System contains four boric acid transfer pumps. Two of these pumps are normally assigned to each unit but valving and piping arrangements allow pumps to be shared such that 3 out of 4 pumps can service either unit. An alternate (not normally used) method of boration is to use the charging pumps taking suction directly from the refueling water storage tank. There are two sources of borated water available to the suction of the charging pumps through two different paths, one from the refueling water storage tank and one from the discharge of the boric acid transfer pumps.

A. The boric acid transfer pumps can deliver the boric acid tank contents (7.0% solution of boric acid) to the charging pumps.

B. The charging pumps can take suction from the volume control tank, the boric acid transfer pumps and the refueling water storage tank. Reference is made to Technical Specification 3.3.

The quantity of boric acid in storage from either the boric acid tanks or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

Approximately 6000 gallons of the 7.0% solution of boric acid are required to meet cold shutdown conditions. Thus, a minimum of 6000 gallons in the boric acid tank is specified. An upper concentration limit of 8.5% boric acid in the tank is specified to maintain solution solubility at the specified low temperature limit of 112°F. For redundancy, two channels of heat tracing are installed on lines normally containing concentrated boric acid solution.

The Boric Acid Tank(s), which are located above the Boron Injection Tank(s), are supplied with level alarms, which would annunciate if a leak in the system occurred.

References

FSAR Section 9.1 Chemical and Volume Control System

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

4. Two charging pumps are operable.

5. Two low head safety injection pumps are operable.

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

7. 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.

8. 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

9. 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

10. 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

11. 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

12. 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.

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. 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.

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.

	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 Heat 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.	Volume Control Tank Level	N.A.	R	N.A.	
17.	Reactor Containment Pressure-CLS	*D	R	M (1)	1) Isolation Valve signal and spray signal
18.	Process and Area Radiation Monitoring System	*D	R	M	
19.	Boric Acid Control	N.A.	R	N.A.	
20.	Containment Sump Level	N.A.	R	N.A.	
21.	Accumulator Level and Pressure	S	R	N.A.	
22.	Containment Pressure-Vacuum Pump System	S	R	N.A.	
23.	Steam Line Pressure	S	R	M	
24.	Turbine First Stage Pressure	S	R	M	
25.	Emergency Plan Radiation Instr.	*M	R	M	

TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
26.	Environmental Radiation Monitors	*M	N.A.	N.A.	TLD Dosimeters
27.	Logic Channel Testing	N.A.	N.A.	M	
28.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	R	
29.	Turbine Trip Setpoint	N.A.	R	R	Stop valve closure or low EH fluid pressure
30.	Seismic Instrumentation	M	R	M	
31.	Reactor Trip Breaker	N.A.	N.A.	M	
32.	Reactor Coolant Pressure	N.A.	R	N.A	
33.	Auxiliary Feedwater				
	a. Steam Generator Water Level Low-Low	S	R	M	
	b. RCP Undervoltage	S	R	M	
	c. S.I.	(All Safety Injection surveillance requirements)			
	d. Station Blackout	N.A.	R	N.A	
	e. Main Feedwater Pump Trip	N.A.	N.A.	R	

S - Each shift M - Monthly
D - Daily P - Prior to each startup if not done previous week
W - Weekly R - Each Refueling Shutdown
NA - Not applicable BW - Every two weeks
SA - Semiannually AP - After each startup if not done previous week
Q - Every 90 effective full power days

*See Specification 4.1D

Channel Description	Check	Calibrate	Test	Remarks
34. Loss of Power				
a. 4.16 KV Emergency Bus undervoltage (Loss of voltage)	N.A.	R	M	
b. 4.16 KV Emergency Bus undervoltage (Degraded voltage)	N.A.	R	M	

TABLE 4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
1.	Reactor Coolant Liquid Samples	Radio-Chemical Analysis (1)	Monthly (5)	
		Gross Activity (2)	5 days/week (5)	9.1
		Tritium Activity	Weekly (5)	9.1
		*Chemistry (CL, F & O_2)	5 days/week	4
		*Boron Concentration	Twice/week	9.1
		$\bar{E}$ Determination	Semiannually (3)	
		DOSE EQUIVALENT I-131	Once/2 weeks (5)	
		Radio-iodine Analysis (including I-131, I-133 & I-135)	Once/4 hours (6) and (7) below	
2.	Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly	6
3.	Boric Acid Tanks	*Boron Concentration	Twice/Week	9.1
4.	Chemical Additive Tank	NaOH Concentration	Monthly	6
5.	Spent Fuel Pit	*Boron Concentration	Monthly	9.5
6.	Secondary Coolant	Fifteen minute degassed b and q activity (4)	Once/72 hours	10.3
		DOSE EQUIVALENT I-131	Monthly (4) Semiannually (8)	
7.	Stack Gas Iodine and Particulate Samples	*I-131 and particlate radioactive releases	Weekly	
8.	Accumulator	Boron Concentration	Monthly	6.2

*See Specification 4.1.D

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By letter dated September 13, 1983, as supplemented October 6, November 30, December 19, 1983, and January 18 and 25, 1984, the Virginia Electric and Power Company (the licensee) requested amendments to the Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, in the form of changes to the Technical Specifications. The proposed changes would reduce the boron concentration in the Boron Injection Tank (BIT). Specifically, the proposed changes to the Technical Specifications will eliminate the minimum boron concentration requirement in the BIT, and reduce the boron concentration requirement in the boric acid system from 11.5 wt% to 7.0 wt%.

Background

Westinghouse incorporated a boron injection tank (BIT) into the Surry Safety Injection (SI) system to mitigate the consequences of postulated steam line break (SLB) events by purging highly concentrated boric acid solution (20,000 ppm B) into the RCS. The licensee has submitted a request for Technical Specification changes including reduction of the BIT boric acid concentration from 11.5% (20,000 ppm B) to 0%, and reduction of the minimum boric acid concentration in the boric acid tanks (BATs) from 11.5% to 7% (12,000 ppm B). The minimum specified BAT temperature would be reduced from 145°F to 112°F, and the BIT temperature specification would be deleted. The licensee's proposed Technical Specification changes include increasing the minimum allowable BAT inventory associated with each unit from 4200 gallons to 6000 gallons. This would preserve the capability for cold shutdown at any time in core life with the most reactive control rod assembly withdrawn from the core.

The licensee has stated that the requested change would reduce maintenance problems and associated personnel radiation exposure (136 man-rem savings) by reducing leakage due to corrosion and decreasing heat tracing circuitry failures. The latter can cause line plugging and flow restrictions as the temperature decreases and precipitation of the concentrated boric acid solution occurs. The potential for corrosion of carbon steel components and supports due to leakage would also be reduced. The physical modifications involved in the proposed change would include cutting the recirculation between the BIT and the BAT, welding the ends closed, and removing the electric power to the recirculation line isolation valve, BIT heaters and recirculation line heat tracing. The BIT would remain in place.

Evaluation

The design basis accident for which the BIT was designed is the Main Steam Line Break (MSLB). The licensee therefore submitted revised analyses which include the MSLB accident with and without offsite power as well as a smaller break equivalent to the capacity of a single steam dump valve or safety valve.

The Surry steam generators are equipped with integral flow restrictors at the generator outlet which serve to reduce the largest effective break area to 1.4 ft^2. Each main steam line has a fast closing trip valve, designed to close in less than 5 seconds, and a non-return valve. These valves prevent blowdown of more than one steam generator even if one valve fails to close.

The energy removal due to the MSLB causes a rapid reduction of reactor coolant system (RCS) temperature and pressure. This results in an increase in reactivity and decrease in shutdown margin. The analysis assumes conservative initial conditions, including hot shutdown with all but the most reactive control rod inserted at end of core life. These assumptions maximize the positive reactivity insertion resulting from cooldown. The single most restrictive failure of the Engineered Safety Features is also assumed, resulting in operation of only one high pressure Safety Injection (SI) pump. The time delay required to sweep unborated water in the BIT and associated SI piping prior to delivery of borated water of 2000 ppm from the RWST has been included in the analysis.

The MSLB analysis was performed by the licensee using the RETRAN computer code. The analyses indicate return to criticality for all 3 cases, with the highest peak heat flux of 23.7% achieved for the MSLB with offsite power available.

In the determination of the critical heat flux at which burnout could occur during a steam line break, the W-3 correlation was used. This correlation is generally considered valid between pressures of 1000 to 2300 psia. The resulting RCS pressures in the three steam line break cases which were reanalyzed were below 1000 psia and ranged from 959 to 733 psia. Although the staff has not approved the use of the W-3 correlation below pressures of 1000 psia, the calculated DNBRs were appreciably higher than the W-3 1.30 limit, and the staff concludes that there is sufficient conservatism in the Surry steam line break calculations to assure that DNB will not occur.

Elimination of the boron concentration requirement in the BIT could affect the containment pressure and temperature response under MSLB accident conditions through changes in the mass and energy release rates. The licensee has performed sensitivity studies to address the impact of reducing the BIT boron concentration on early MSLB energy release, and has concluded that the current equipment qualification temperature envelopes for the Surry plants are adequate. Since LOCA conditions dominate the containment functional design considerations, the licensee used the LOCA temperature profiles for post-accident equipment qualification in lieu of MSLB temperature profiles. We have also made comparisons with similar reviews for the Beaver Valley and

North Anna plants. The boron concentration reduction programs at these plants were previously found acceptable by the staff. Based on a review of the information submitted by the licensee, and because of the similarity of the licensee's request to other staff actions, we conclude that the licensee's proposal to eliminate the minimum boron concentration requirement in the BIT will not adversely affect the containment functional performance.

The licensee evaluated three scenarios with an MSLB and its associated cooling of the primary system. For each, the licensee calculated a worst-case return-to-power transient. Of these, the scenario with the greatest peak power after scram was that which included assumptions of zero power at the break, the availability of offsite power, and the largest possible break. (Other licensee assumptions to make the scenarios worst-case with respect to return to power included a minimum reactivity shutdown margin, end-of-life core moderator temperature coefficient, the highest-worth control rod assembly stuck in its fully withdrawn position, and the failure of one high-pressure coolant injection pump.) We also assumed coolant iodine equilibrium and spiking consistent with SRP section 15.1.5. This scenario is, however, not the worst case with respect to radiological consequences.

For example, the zero-power initial condition assumption implies that there is negligible decay heat, thus easing the long-term plant cooldown. Also, the availability of offsite power increases the chances of better mitigating the accident by using the condenser to retain iodine from a potential leak of primary coolant to the secondary side of the unaffected steam generators. We were concerned about the potential for prolonged plant cool-down and primary-to-secondary leakage caused by additional energy from the return to power, which might be in addition to normal decay heat. However, we found that significant return-to-power would not cause energy input to the primary system that would be additive to decay heat, since the large return-to-power cases involved zero power initial conditions. Therefore, we determined that the worst-case accident (assuming there is no additional fuel failure) is for full power initial conditions, together with loss of offsite power. Additionally, we assumed no further fuel or cladding defects because of the finding that the worst-case return to power reported by the licensee would not result in further fuel or cladding failure (beyond the assumed minor cladding defects associated with the assumed iodine spike).

Further, the proposed amendment does not affect the validity of the original staff MSLB dose evaluation in the Surry SER. A calculation was performed, using current methods, to confirm that the postulated MSLB doses are within the dose guidelines of 10 CFR 100.11 and appropriate fractions thereof, as defined in the acceptance criteria of SRP 15.1.5. We have determined that the proposed amendment does not exceed or detrimentally affect our radiological consequence guidelines.

The calculated MSLB doses are shown in Table 1, for both the 0-2 hour dose at the Exclusion Area Boundary and for the duration of the accident (judged to be 8 hours) at the outer boundary of the Low Population Zone. The assumptions are given in Table 2. The acceptance criterion for the preaccident spike case is 100% of 10 CFR 100.11 guidelines, or 300 rems thyroid and 25 rems whole body; for the concomitant spike case, it is 10%, or 30 rems thyroid and 2.5 rems whole body.

The licensee has stated that lowering the BIT boric acid concentration to 0 ppm eliminates the need to maintain BIT heaters and heat tracing on the associated SI piping and recirculation lines. We requested information from the licensee on whether the normally stagnant section of the SI piping between the charging pump normal discharge piping and the closed isolation valves upstream of the BIT can contain concentrated boric acid in the event of operator error or equipment failure (e.g., valve leakage), and, if this could occur, to explain how precipitation of boric acid and consequent possible pipe blockage would be avoided.

The licensee responded that the charging headers and stagnant SI piping upstream of the BIT inlet valves are not currently heat traced, since they are located inside buildings. Operating experience has indicated that boron precipitation in the charging header and stagnant lines has not been a problem. The stagnant SI piping between the main charging header and the BIT inlet valves is flushed monthly in accordance with Technical Specification 4.1.E. Also, the licensee has not deleted the Technical Specifications requirement to have two channels of heat tracing available for the SI flow paths. We conclude that the licensee's response on this subject is acceptable.

Several letters have been received from the licensee since these requests for amendments were published in the Federal Register on November 22, 1983. These letters are dated November 30 and December 19, 1983, and January 18 and 25, 1984. None of these letters changed the Technical Specifications proposed by the application dated September 13, 1983, nor the substance of the application in a significant manner and were of a nature of providing additional clarification and details to the staff. These letters are summarized in the following paragraphs.

The November 30, 1983 letter responded to staff questions related to the containment response following a postulated design basis main steam line rupture with a reduced boron concentration. The licensee compares the Surry analysis to that performed for the Beaver Valley Power Station and concludes that the Beaver Valley calculations are bounding.

The December 19, 1983 response provides additional details of the reactivity feedback model and mixing coefficients utilized in the MSLB analysis, clarification of heat tracing requirements and discussion of offsite doses.

The January 18, 1984 letter provides an estimate of the man-rem reduction that the proposed change would effect. This letter was provided for the interest of the staff and was not the basis for the review.

The January 25, 1984 letter provides a tabulation of data of MSLB Accident Statepoints which confirm that DNBR is well above 1.3.

An insignificant change was made to Table 4.1-1 in that the line item related to Boron Injection Tank Level has been deleted instead of being shown as not applicable. (This was not in the licensee's submittal but was discussed by telephone.)

Based on our review, we conclude that the proposed Technical Specification changes will not result in unacceptable consequences in the event of the design basis accident, and are acceptable.

Environmental Consideration

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.

Conclusion

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

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Table 1 Radiological Consequences of a Postulated Main Steam Line Break

Description	Thyroid dose, rems	Whole body dose, rems
Pre-accident iodine spike*		
0-2 hour, Exclusion Area Boundary	5.2	less than 0.1
0-8 hour, outer boundary of Low Population Zone	0.4	less than 0.1
Concomitant iodine spike (caused by accident)		
0-2 hour, Exclusion Area Boundary	7.8	less than 0.1
0-8 hour, outer boundary of Low Population Zone	2.3	less than 0.1

*That these doses are lower than the concomitant iodine spike doses is atypical compared to other plants, and is caused by the low Technical Specifications on the short-term maximum coolant iodine concentration.

Table 2 Assumptions Used in Estimating Doses From a Main Steam Line Break

1. The break occurs on a main steam line between the containment penetration and the main steam isolation valve. The affected steam generator boils dry.

2. During the rapid boil-off, all activity in the affected steam generator is released to the environment. The secondary side iodine concentration was assumed to be 0.1 microcurie dose equivalent (DE) I-131 per cc. The steam generator liquid volume was 1690 cubic feet.

3. Additional activity is released via a primary-secondary leak. This is assumed to be at the maximum allowed by technical specifications, which for Surry is .347 gallons per minute (gpm) to any one steam generator (assumed to be the affected generator), and a total of 1. gpm to all the steam generators.

4. Pre-accident spike only case: before the accident, an iodine spike has ocurred which brings the primary coolant activity up to the technical specification limit for 48-hour operation, 10 microcuries per cc DE I-131.

5. Iodine spike caused by accident case: before the accident, the primary coolant activity is at the technical specification limit for long-term operation, 1.0 microcurie per cc DE I-131. With the start of the accident an iodine spike begins, which releases an additional 7400 Ci/hr, for 4 hours, from the core to the coolant.

6. Because the time that the steam generator tube bundles in the affected steam generator are fully covered is small compared to the total duration of the accident, it is assumed (1) that the primary-to-secondary leak to the affected steam generator occurs entirely in a dry section, and (2) that all the activity in the leaked coolant is released to the environment. For the unaffected steam generators, it was assumed that the condenser would be unavailable, and that steam release to the environment would take place. It was assumed, however, that only 1% of the iodine in the primary coolant leaked to the unaffected steam generators would be released to the environment.

7. Primary-to-secondary leaks become negligible after 8 hours, and all releases to the environment cease.

8. All the noble gases in the leaked coolant, with a concentration equal to the technical specification limit, are released to the environment.