Docket Nos. 50-280
and 50-281

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Virginia Electric & Power Company
ATTN: Mr. Stanley Ragone
Senior Vice President
Post Office Box 26666
Richmond, Virginia 23261

Gentlemen:

The Commission has issued the enclosed Amendments No. 8 to Facility Licenses No. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. The amendments include Change No. 23 to your Technical Specifications for each license and are in response to your requests dated September 6, 1974 and September 24, 1974.

The amendments revise the provisions in the Technical Specifications relating to 20 miscellaneous items. Seven of the revisions involve the Administrative Controls section of the Technical Specifications, another six correct errors of various types, and the remaining seven revisions involve minor changes to bring the Technical Specifications in line with the "as-built" plant, to clarify the specifications or to make the specifications consistent with the Final Safety Analysis Report (FSAR).

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

Sincerely,

V L. Rooney
for

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 8 to DPR-32
2. Amendment No. 8 to DPR-37
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures: See next page

OFFICE ➤	DRL:ORB#1	TR ADRS	OELD	DRL:ORB#1	DRL:OR	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 8
License No. DPR-32

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

   A. The applications for amendment by Virginia Electric & Power Company (the licensee) dated September 6, 1974 and September 24, 1974 comply 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 applications 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; and

   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.

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

"3.B Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 23."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 23 to the
Technical Specifications

Date of Issuance: July 22, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 8

CHANGE NO. 23 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

Revise Appendix A as follows:

Remove Page	Insert New Page
ii	ii
1.0-1	1.0-1
1.0-2	1.0-2
3.1-21	3.1-21
3.3-3	3.3-3
3.6-4	3.6-4
3.17-1	3.17-1
4.1-1	4.1-1
4.1-6 (table)	4.1-6 (table)
4.1-7 (table)	4.1-7 (table)
4.1-8 (table)	4.1-8 (table)
4.1-10 (table)	4.1-10 (table)
4.4-3	4.4-3
4.9-1	4.9-1
6.1-1	6.1-1
6.1-6	6.1-6
6.1-7	6.1-7
6.1-8	6.1-8
6.1-10	6.1-10
Fig. 6.1-3	Fig. 6.1-3 (Blank)
6.2-1	6.2-1
6.4-7	6.4-7a
	6.4-7b
	6.4-7c
6.4-8 (table)	6.4-8 (table)
6.6-9	6.6-9
6.6-10	6.6-10

1.0 DEFINITIONS

The following frequently used terms are defined for the uniform interpretation of the specifications.

A. Rated Power

A steady state reactor core heat output of 2441 MWt.

B. Thermal Power

The total core heat transferred from the fuel to the coolant.

C. Reactor Operation

1. Refueling Shutdown Condition

When the reactor is subcritical by at least 10% $\Delta k/k$ and T_{avg} is $\leq 140^{o}F$ and fuel is scheduled to be moved to or from the reactor core.

2. Cold Shutdown Condition

When the reactor is subcritical by at least 1% $\Delta k/k$ and T_{avg} is $\leq 200^{o}F$.

3. Intermediate Shutdown Condition

When the reactor is subcritical by an amount greater than or equal to the margin as specified in Technical Specification Figure 3.12-7 23
and $200^{o}F < T_{avg} < 547^{o}F$.

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4. Hot Shutdown Condition

When the reactor is subcritical by an amount greater than or equal to the margin specified in Technical Specification Figure 3.12.7 | 23 and T_{avg} is $\geq 547^{o}F$.

5. Reactor Critical

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

6. Power Operation

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

7. Refueling Operation

A Any operation involving movement of core components when the vessel head is unbolted or removed.

D. Operable

A system or component is operable when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the limiting conditions for operation defined in Section 3, and (2) it has been tested periodically in accordance with Section 4 and meets its performance requirements.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is below 250°F:

Contaminant	Normal Concentration (PPM)	Transient not to exceed 24 hours (PPM)
a. Oxygen	Saturated	Saturated
b. Chloride	0.15	1.5
c. Fluoride	0.15	1.5

If the limits above are exceeded, the reactor shall be immediately brought to the cold shutdown condition and the cause of the out-of-specification condition shall be ascertained and corrected.

5. For the purposes of correcting the contaminant concentrations to meet technical specifications 3.1.F.1 and 3.1.F.4 above, increase in coolant temperature consistent with operation of primary coolant pumps for a short period of time to assure mixing of the coolant shall be permitted. This increase in temperature to assure mixing shall in no case cause the coolant temperature to exceed 250°F. 23

6. If more than one contaminant or contaminants transient, which results in contaminant levels exceeding any of the normal steady state operation limits specified in 3.1.F.1 or 3.1.F.4, is experienced in any seven consecutive day period, the reactor shall be placed in a cold shutdown condition until the cause of the out-of-specification operation is ascertained and corrected.

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. 23

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 shut down.

4. Any one valve in the Safety Injection System may be inoperable provided repairs are initiated immediately and are completed within 24 hours. Prior to initiating repairs, all automatic valves in the redundant system shall be tested to demonstrate operability.

5. One channel of heat tracing may be inoperable for a period not to exceed 24 hours, provided immediate attention is directed to making repairs.

6. One charging pump component cooling water pump or one charging pump service water pump may be out of service provided the pump is

coolant loop operation. 23

The availability of the auxiliary feedwater pumps, the protected condensate storage tank, and the main steam line safety valves adequately assures that sufficient residual heat removal capability will be available when required.

The limit on steam generator secondary side iodine - 131 activity is based on limiting inhalation thyroid dose at the site boundary to 1.5 rem after a postulated accident that would result in the release of the entire contents of a unit's steam generators to the atmosphere. In this accident, with the halogen inventories in the steam generator being at equilibrium values, I-131 would contribute 75 percent of the resultant thyroid dose at the site boundary; the remaining 25 percent of the dose is from other isotopes of iodine. In the analysis, one-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plate out and retention in water droplets.

The inhalation thyroid dose at the site boundary is given by:

$$\text{Dose (Rem)} = \frac{(C)\ (\chi/Q)\ (D_\infty/A_\tau)\ (B.R.)}{(.75)\ (P.F.)}$$

where: C = steam generator I-131 activity (curies)

$\chi/Q = 8.14 \times 10^{-4}$ sec/m^3

$D_\infty/A_\tau = 1.48 \times 10^{6}$ rem/Ci for I-131

B.R. = breathing rate, 3.47×10^{-4} m^3/sec.

from TID 14844

P.F. = plating factor, 10

Assuming the postulated accident, the resultant thyroid dose is 1.5 rem.

3.17 LOOP STOP VALVE OPERATION

Applicability

Applies to the operation of the Loop Stop Valves.

Objective

To specify those limiting conditions for operation of the Loop Stop Valves which must be met to ensure safe reactor operation.

Specifications

1. Whenever a reactor coolant loop is isolated, the boron concentration in the isolated loop shall be maintained at a value greater than or equal (± 2% analytical error) to the boron concentration in the active loops. The boron concentration in an isolated loop shall be measured and logged at least 5 days per week. 23

2. Whenever startup of an isolated reactor coolant loop is initiated, the following conditions shall be met:

   a. All the channels, including redundant channels, of the Loop Stop Valve Interlock System of the isolated loop are operable. In the event htis condition is not satisfied, the loop must remain isolated.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

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

Objective

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

Specification

A. Calibration, testing, and checking of instrumentation channels shall be performed as detailed in Table 4.1-1.

B. Equipment tests shall be conducted as detailed in Table 4.1-2A.

C. Sampling tests shall be conducted as detailed in Table 4.1-2B.

D. Whenever containment integrity is not required, only the asterisked items in Table 4.1-1 and 4.1-2A and 4.1-2B are applicable. 23

E. Flushing of sensitized stainless steel pipe sections shall be conducted as detailed in TS Table 4.1-3A and 4.1-3B.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND

TESTS OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S M(3)	D (1) Q (3)	BW(2)	1) Against a heat balance standard 2) Signal to ΔT; bistable action (permissive, rod stop, strips) 3) Upper and lower chambers for symetric offset by means of the moveable incore detector system.
2.	Nuclear Intermediate Range	*S(1)	N.A.	P(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	*S (1)	N.A.	P(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	*S	R	BW(1) BW(2)	1) Overtemperature - ΔT 2) Overpower - ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure(High & Low)	S	R	M	
8.	4 Kv Voltage & Frequency	S	R	M	Reactor protection circuits only
9.	Analog Rod Position	*S(1,2) (4)	R	M(3)	1) With step counters 2) Each six inches of rod motion when data logger is out of service 3) Rod bottom bistable action 4) NA When reactor is in cold shut-down

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TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S (1,2)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	*D	R	N.A.	
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	Boron Injection Tank Level	W	N.A.	N.A.	23
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 Sump 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	

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TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
25.	Turbine First Stage Pressure	S	R	M	
26.	Emergency Plan Radiation Instruments	*M	R	M	
27.	Environmental Radiation Monitors	*M	N.A.	N.A.	TLD Dosimeters
28.	Logic Channel Testing	N.A.	N.A.	M	
29.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	R	
30.	Turbine Trip Set Point	N.A.	R	R	Stop valve closure or low EH fluid pressure
31.	Seismic Instrumentation	M	SA	M	
32.	Reactor Trip Breaker	N.A.	N.A.	M	

S - Each Shift
D - Daily
W - Weekly
NA - Not applicable
SA - Semiannually
Q - Every 90 effective full power days

M - Monthly
P - Prior to each startup if not done previous week
R - Each Refueling Shutdown
BW - Every two weeks
AP - After each startup if not done previous week

* See Specification 4.1D

23

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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[6]		
	Gross Activity[2]	5 days/week[6]	9.1	
	Tritium Activity	Weekly[6]	9.1	
	*Chemistry(Cl, $F\&O_2$)	5 days/week	4	23
	*Boron Concentration	Twice/week	9.1	
	$\overline{E}$ Determination	Semiannually[3]		
2. Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly	6	
3. Boric Acid Tanks	*Boron Concentration	Twice/week	9.1	
4. Boron Injection Tank	Boron Concentration	Twice/week	6	
5. Chemical Additive Tank	NaOH Concentration	Monthly	6	
6. Spent Fuel Pit	*Boron Concentration	Monthly	9.5	
7. Secondary Coolant	Fifteen minute degassed β and γ activity[4]	Weekly	10.3	
8. Stack Gas Iodine and Particulate Samples	*I-131 and particulate radioactive releases[5]	Weekly		
9. 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.

(3) $\overline{E}$ determination will be started when the gross gamma degassed activity of radionuclides with half-lives greater than 30 minutes analysis indicates $\geq$ 10μCi/cc.

(4) If the fifteen minute degassed beta and gamma activity is 10% of that given in Specification 3.6.C, an I-131 analysis will be performed.

(5) If the activity of the samples is 10% or greater of that given in Specification 3.11.B.1. the frequency shall be increased to daily.

(6) When reactor is critical.

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Basis

The leaktightness testing of all liner welds was performed during construction by welding a structural steel test channel over each weld seam and performing soap bubble and halogen leak tests.

The containment is designed for a maximum pressure of 45 psig. The containment is maintained at a subatmospheric air partial pressure which varies between 9 psia and 11 psia depending upon the cooldown capability of the Engineered Safeguards and is not expected to rise above 45 psig for any postulated loss-of-coolant accident. 23

All loss-of-coolant accident evaluations have been based on an integrated containment leakage rate not to exceed 0.1 percent of containment volume per 24 hr.

The above specification satisfies the conditions of 10 CFR 50.54(0) which states that primary reactor containments shall meet the containment leakage test requirements set forth in Appendix J.

References

FSAR Section 5.4 Design Evaluation of Containment Tests and Inspections of Containment

FSAR Sections 7.5.1 Design Bases of Engineered Safeguards Instrumentation

FSAR Section 14.5 Loss-of-Coolant Accident

10 CFR 50 Appendix J (Proposed), "Reactor Containment Leakage Testing for Water Cooled Power Reactors," as published in the Federal Register, Volume 36, No. 167, August 27, 1971.

4.9 EFFLUENT SAMPLING AND RADIATION MONITORING SYSTEM

Applicability

Applies to the periodic monitoring and recording of radioactive effluents.

Objective

To ascertain that radioactive releases are maintained as low as practicable and within the limits set forth in 10CFR20.

Specification

A. Procedures shall be developed and used, and equipment which has been installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences, shall be maintained and used to keep levels of radioactive materials in effluents released to unrestricted areas as low as practicable.

B. All effluents to be discharged to the atmosphere from the waste gas decay tanks of the Gaseous Waste Disposal System shall be sampled prior to release via the process vent. Effluent from the Liquid Waste Disposal System shall be continuously monitored by the circulating water discharge tunnel monitor and shall be sampled prior to being discharged into the circulating water discharge tunnel. 23

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, SAFETY AND OPERATION REVIEW

Specification

A. The Station Manager shall be responsible for the safe operation of the facility. The Station Manager shall report to the Manager Production Operation and Maintenance. The relationship between this Manager and other levels of company management is shown in TS Figures 6.1-1 and 6.1-2. 23

B. The station organization shall conform to the chart as shown in TS Figure 6.1-4.

1. Qualifications with regard to education and experience and the technical specialties of key supervisory personnel will meet the minimum acceptable levels described in Regulatory Guide No. 1.8 "Personnel Selection and Training," dated March 10, 1971.

   The key supervisory personnel are as follows:

   a) Manager
   b) Superintendent-Station Operations
   c) Operating Supervisor
   d) Supervisor-Electrical Maintenance
   e) Supervisor-Mechanical Maintenance
   f) Supervisor-Engineering Services
   g) Chemistry and Health Physics Supervisor
   h) Shift Supervisor

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Safety and Operating Committee and copies shall be sent to the Manager Production Operation and Maintenance and to all members of the Station and System Nuclear Safety and Operating Committees.

h. Procedures

Written administrative procedures for committee operation shall be prepared and maintained describing the method of submission, and the content of presentations to the committee, provisions for the use of subcommittees; review and approval by members of written committee evaluations and recommendations; the distributions of minutes; and, such other matters as may be appropriate.

2. System Nuclear Safety and Operating Committee

a. Membership

1. Chairman and Vice Chairman appointed by name by the Senior Vice President-Power, which may be an individual listed in Item 2.

2. Seven members of the Power Department system office staff who are experienced in utility operation and procedures:

Manager - Production Operation and Maintenance
Manager - Licensing and Quality Assurance
Manager - Power Station Engineering 23
Design Engineer - Power Station Engineering

Director - Quality Assurance 23

Supervisor - Nuclear Fuel Design and Operation

Supervisor - Nuclear Operation

3. Manager of each nuclear generating station operating on the Virginia Electric and Power Company system, or his designee. In matters or consideration of proposals pertinent to a particular station, the Manager of this station shall serve as a non-voting member of the committee. In matters pertaining to other stations, Station Managers will serve as voting members.

4. At least one qualified non-company affiliated technical consultant. Duly appointed consultant members shall have equal vote with permanent members of the Committee.

b. Qualifications

The minimum qualifications of the Company members of the System Nuclear Safety and Operating Committee will be:

an engineering graduate or equivalent with combined nuclear and conventional experience in power station design and/or operation of eight years, with at least two years involving the direction of nuclear operations or design activity.

c. Consultants

The committee shall have the authority to call technically

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qualified personnel from within the Virginia Electric and Power Company organization or from any other consultant source.

d. Quorum: Either the Chairman or Vice Chairman and two thirds of the other members shall constitute a quorum.

e. Meeting frequency: As required by the Chairman but not less than quarterly.

f. Responsibilities

1. Review proposed changes to the operating license including Technical Specifications.

2. Review minutes of meeting of the Station Nuclear Safety and Operating Committee(s) to determine if matters considered by that Committee involve "unreviewed safety questions" as defined in 10 CFR 50.59. 23

3. Review matters including proposed changes or modifications by systems or equipment having safety significance referred to it by the Station Nuclear Safety and Operating Committee or by the Station Manager.

4. Conduct periodic review of station operations.

5. Review all reported instances of departure from Technical Specifications limits and report findings and recommendations to prevent recurrence to the Vice President-Power Supply and Production Operations. 23

Copies of the minutes shall be forwarded to the Senior Vice President-Power, Vice President-Power Supply and Production Operations, all members of the committee and any others that the Chairman may designate. 23

i. Procedures

Written administrative procedures for committee operation shall be maintained describing the method of submission and the content of presentations to the committee; provisions for use of subcommittee evaluations and recommendations; distribution of minutes; and, such other matters as may be appropriate.

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6.2 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE IN STATION OPERATION

Specification

A. Any abnormal occurrence shall be reported immediately to and promptly reviewed by the Chairman-Station Nuclear Safety and Operating Committee or his designee, Manager-Production Operations and Maintenance and the Chairman-System Nuclear Safety and Operating Committee. Prompt corrective action shall be taken to correct the anomaly.

B. A qualified individual shall prepare a written report for each abnormal occurrence for submittal to the NRC. This report shall include an evaluation of the cause of the occurrence and also recommendations for appropriate action to prevent or reduce the probability of a recurrence. 23

C. Copies of all such reports shall be submitted to the Superintendent-Station Operations, the Station Manager, who also serves as the Chairman-Station Nuclear Safety and Operating Committee, Manager-Production Operations and Maintenance, and to the Chairman-System Nuclear Safety and Operating Committee for review and approval of any recommendation.

D. A duly authorized executive of the Company, or designee, shall report the circumstances of any abnormal occurrence to the NRC as specified in Section 6.6.B.1 of these Specifications. 23

D. All procedures described in A and B above shall be followed.

E. Temporary changes to procedures described in A and B above which do not change the intent of the original procedure may be made, provided such changes are approved prior to implementation by the person designated below based on the type of procedure to be changed:

1.	Administrative	Station Manager
2.	Abnormal	Shift Supervisor
3.	Annunciator	Shift Supervisor
4.	Chemistry and Health Physics	Health Physist, Chemist
5.	Emergency	Shift Supervisor
6.	Electrical Maintenance	Electrical Foreman
7.	Mechanical Maintenance	Mechanical Foreman
8.	Operating	Shift Supervisor
9.	Periodic Test	Cognizant Supervisor
10.	Start-up Test	Supervisor-Engineering Services
11.	Special Test	Supervisor-Engineering Services
12.	Quality Control	Quality Control Engineer

23

Such changes will be documented and subsequently reviewed by the Station Nuclear Safety and Operating Committee and approved by the Station manager within seven days.

F. Temporary changes to procedures described in A and B above which change the intent of the original procedure may be made, provided such changes are approved prior to implementation by the person designated below based on the type of procedure to be changed.

1.	Administrative	Station Manager
2.	Abnormal	Operating Supervisor or Superintendent Station Operations
3.	Annunciator	Operating Supervisor or Superintendent Station Operations
4.	Chemistry and Health Physics	Supervisor-Chemistry and Health Physics
5.	Emergency	Operating Supervisor or Superintendent Station Operations
6.	Maintenance	Supervisor-Mechanical Maintenance Supervisor-Electrical Maintenance Instrument Supervisor
7.	Operating	Operating Supervisor Superintendent-Station Operations
8.	Periodic Test	Supervisor-Engineering Services
9.	Startup Test	Supervisor-Engineering Services
10.	Special Test	Supervisor-Engineering Services
11.	Quality Control	Quality Control Engineer

23

Such changes will be documented and subsequently reviewed by the Station Nuclear Safety and Operating Committee and approved by the Station Manager.

G. In cases of emergency, operations personnel shall be authorized to depart from approved procedures where necessary to prevent injury to personnel or damage to the facility. Such changes shall be documented and reviewed by the Station Nuclear Safety and Operating Committee and approved by the Station Manager.

H. Practice of site evacuation exercises shall be conducted annually, following emergency procedures and including a check of communications with off-site report groups. An annual review of the <u>Emergency Plan</u> will be performed.

I. The industrial security program which has been established for the station shall be implemented, and appropriate investigation and/or corrective action shall be taken if the provisions of the program are violated. An annual review of the program shall be performed. 23

TABLE TS 6.4-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES 1/	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT NOBLE GASES AND TRITIUM OXIDE 3/	23
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 6/	NP	5	
Facepiece, full 6/	NP	100	
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Air Line respirator			
Facepiece, half-mask	CF	100	
Facepiece, full	CF	1,000	
Facepiece, full 6/	D	100	7
Facepiece, full	PD	1,000	
Hood	CF	5/	
Suit	CF	5/	
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full 6/	D	100	7
Facepiece, full	PD	1,000	
Facepiece, full	R	1,000	
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-sypplying respirator		Protection factors for type and mode of operation as listed above.	

B. Non-Routine Reports

1. Abnormal Occurrence Reports - A notification shall be made within 24 hours by telephone and telegraph to the Director of the Regional Regulatory Operations Office (cc to the Director of Licensing) followed by a written report within 15 days to the Director of the Regional Regulatory Operations Office in the event of an abnormal occurrence. The written abnormal occurrence report should follow the format presented in Regulatory Guide 1.16. 23

2. Reporting of Unusual Safety Related Events

A written report shall be forwarded within 30 days to the Director of the Regional Regulatory Operations Office in the event of an unusual safety related event.

Unusual safety related events are defined in Section 1 of these Technical Specifications.

3. Radioactive Effluents

If the experienced rate of release of radioactive materials in liquid and gaseous wastes, when averaged over any 48 hour period is such that the concentrations of radionuclides at the site boundary exceed 4% of the limits in Specification 3.11-A.1 or 3.11-B-1, a written report shall be submitted to the Deputy Director for Reactor Projects, Directorate of Licensing and to the Director,Region II Directorate of Regulatory Operations within 30 days following the release. This report shall identify the cause(s) of the activity release and describe the proposed program of action to reduce such release(s). | 23

C. Inservice Inspection Reports

Notification shall be made to the Director, Region II Directorate of Regulatory Operations at least 60 days before scheduled refueling periods. This notification shall include the details of the inservice inspection planned for such period, including a detailed schedule identifying specific welds to be inspected and identifying the specific techniques to be employed.

Within 90 days after completion of inservice inspection activities, a written report shall be forwarded to the Deputy Director for Reactor Projects, Directorate of Licensing, and to the Director, Region II, Directorate of Regulatory Operations covering the scope of the examinations conducted and the results thereof. In addition, a preliminary report of the findings of the inservice inspection activities shall be filed with the Deputy Director for Reactor Projects, Directorate of Licensing, and the Director, Region II Directorate of Regulatory Operation before the facility is returned to power.

D. Special Reports

a. Startup Report - A summary report of unit startup and power escalation testing and the evaluation of the results from these test programs shall be submitted when a unit is initially placed

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 8
License No. DPR-37

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

   A. The applications for amendment by Virginia Electric & Power Company (the licensee) dated September 6, 1974 and September 24, 1974 comply 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 applications 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; and

   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.

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

"3.B Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 23 ."

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

FOR THE NUCLEAR REGULATORY COMMISSION

R. A. Purple, for

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 23 to the
Technical Specifications

Date of Issuance: July 22, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 8

CHANGE NO. 23 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Revise Appendix A as follows:

Remove Page	Insert New Page
ii	ii
1.0-1	1.0-1
1.0-2	1.0-2
3.1-21	3.1-21
3.3-3	3.3-3
3.6-4	3.6-4
3.17-1	3.17-1
4.1-1	4.1-1
4.1-6 (table)	4.1-6 (table)
4.1-7 (table)	4.1-7 (table)
4.1-8 (table)	4.1-8 (table)
4.1-10 (table)	4.1-10 (table)
4.4-3	4.4-3
4.9-1	4.9-1
6.1-1	6.1-1
6.1-6	6.1-6
6.1-7	6.1-7
6.1-8	6.1-8
6.1-10	6.1-10
Fig. 6.1-3	Fig. 6.1-3 (Blank)
6.2-1	6.2-1
6.4-7	6.4-7a
	6.4-7b
	6.4-7c
6.4-8 (table)	6.4-8 (table)
6.6-9	6.6-9
6.6-10	6.6-10

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1.0 DEFINITIONS

The following frequently used terms are defined for the uniform interpretation of the specifications.

A. Rated Power

A steady state reactor core heat output of 2441 MWt.

B. Thermal Power

The total core heat transferred from the fuel to the coolant.

C. Reactor Operation

1. Refueling Shutdown Condition

When the reactor is subcritical by at least 10% $\Delta k/k$ and T_{avg} is $\leq 140^{o}F$ and fuel is scheduled to be moved to or from the reactor core.

2. Cold Shutdown Condition

When the reactor is subcritical by at least 1% $\Delta k/k$ and T_{avg} is $\leq 200^{o}F$.

3. Intermediate Shutdown Condition

When the reactor is subcritical by an amount greater than or equal to the margin as specified in Technical Specification Figure 3.12-7 and $200^{o}F < T_{avg} < 547^{o}F$. 23

4. Hot Shutdown Condition

When the reactor is subcritical by an amount greater than or equal to the margin specified in Technical Specification Figure 3.12.7 and T_{avg} is $\geq 547^oF$. 23

5. Reactor Critical

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

6. Power Operation

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

7. Refueling Operation

A Any operation involving movement of core components when the vessel head is unbolted or removed.

D. Operable

A system or component is operable when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the limiting conditions for operation defined in Section 3, and (2) it has been tested periodically in accordance with Section 4 and meets its performance requirements.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is below 250°F:

Contaminant	Normal Concentration (PPM)	Transient not to exceed 24 hours (PPM)
a. Oxygen	Saturated	Saturated
b. Chloride	0.15	1.5
c. Fluoride	0.15	1.5

If the limits above are exceeded, the reactor shall be immediately brought to the cold shutdown condition and the cause of the out-of-specification condition shall be ascertained and corrected.

5. For the purposes of correcting the contaminant concentrations to meet technical specifications 3.1.F.1 and 3.1.F.4 above, increase in coolant temperature consistent with operation of primary coolant pumps for a short 23 period of time to assure mixing of the coolant shall be permitted. This increase in temperature to assure mixing shall in no case cause the coolant temperature to exceed 250°F.

6. If more than one contaminant or contaminants transient, which results in contaminant levels exceeding any of the normal steady state operation limits specified in 3.1.F.1 or 3.1.F.4, is experienced in any seven consecutive day period, the reactor shall be placed in a cold shutdown condition until the cause of the out-of-specification operation is ascertained and corrected.

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. 23

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 shut down.

4. Any one valve in the Safety Injection System may be inoperable provided repairs are initiated immediately and are completed within 24 hours. Prior to initiating repairs, all automatic valves in the redundant system shall be tested to demonstrate operability.

5. One channel of heat tracing may be inoperable for a period not to exceed 24 hours, provided immediate attention is directed to making repairs.

6. One charging pump component cooling water pump or one charging pump service water pump may be out of service provided the pump is

coolant loop operation. 23

The availability of the auxiliary feedwater pumps, the protected condensate storage tank, and the main steam line safety valves adequately assures that sufficient residual heat removal capability will be available when required.

The limit on steam generator secondary side iodine - 131 activity is based on limiting inhalation thyroid dose at the site boundary to 1.5 rem after a postulated accident that would result in the release of the entire contents of a unit's steam generators to the atmosphere. In this accident, with the halogen inventories in the steam generator being at equilibrium values, I-131 would contribute 75 percent of the resultant thyroid dose at the site boundary; the remaining 25 percent of the dose is from other isotopes of iodine. In the analysis, one-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plate out and retention in water droplets.

The inhalation thyroid dose at the site boundary is given by:

$$\text{Dose (Rem)} = \frac{(C)\ (\chi/Q)\ (D_\infty/A_\tau)\ (B.R.)}{(.75)\ (P.F.)}$$

where: C = steam generator I-131 activity (curies)

$\chi/Q = 8.14 \times 10^{-4}$ sec/m^3

$D_\infty/A_\tau = 1.48 \times 10^6$ rem/Ci for I-131

B.R. = breathing rate, 3.47×10^{-4} m^3/sec.

from TID 14844

P.F. = plating factor, 10

Assuming the postulated accident, the resultant thyroid dose is 1.5 rem.

3.17 LOOP STOP VALVE OPERATION

Applicability

Applies to the operation of the Loop Stop Valves.

Objective

To specify those limiting conditions for operation of the Loop Stop Valves which must be met to ensure safe reactor operation.

Specifications

1. Whenever a reactor coolant loop is isolated, the boron concentration in the isolated loop shall be maintained at a value greater than or equal (± 2% analytical error) to the boron concentration in the active loops. The boron concentration in an isolated loop shall be measured and logged at least 5 days per week. 23

2. Whenever startup of an isolated reactor coolant loop is initiated, the following conditions shall be met:

   a. All the channels, including redundant channels, of the Loop Stop Valve Interlock System of the isolated loop are operable. In the event htis condition is not satisfied, the loop must remain isolated.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

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

Objective

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

Specification

A. Calibration, testing, and checking of instrumentation channels shall be performed as detailed in Table 4.1-1.

B. Equipment tests shall be conducted as detailed in Table 4.1-2A.

C. Sampling tests shall be conducted as detailed in Table 4.1-2B.

D. Whenever containment integrity is not required, only the asterisked items in Table 4.1-1 and 4.1-2A and 4.1-2B are applicable. 23

E. Flushing of sensitized stainless steel pipe sections shall be conducted as detailed in TS Table 4.1-3A and 4.1-3B.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND

TESTS OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S M(3)	D (1) Q (3)	BW(2)	1) Against a heat balance standard 2) Signal to ΔT; bistable action (permissive, rod stop, strips) 3) Upper and lower chambers for symetric offset by means of the moveable incore detector system.
2.	Nuclear Intermediate Range	*S(1)	N.A.	P(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	*S (1)	N.A.	P(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	*S	R	BW(1) BW(2)	1) Overtemperature - ΔT 2) Overpower - ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure(High & Low)	S	R	M	
8.	4 Kv Voltage & Frequency	S	R	M	Reactor protection circuits only
9.	Analog Rod Position	*S(1,2) (4)	R	M(3)	1) With step counters 2) Each six inches of rod motion when data logger is out of service 3) Rod bottom bistable action 4) NA When reactor is in cold shut-down

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TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S (1,2)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	*D	R	N.A.	
15.	Refueling Water Storage Tank Level	W	R	N.A.	
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 Sump 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	

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TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
25.	Turbine First Stage Pressure	S	R	M	
26.	Emergency Plan Radiation Instruments	*M	R	M	
27.	Environmental Radiation Monitors	*M	N.A.	N.A.	TLD Dosimeters
28.	Logic Channel Testing	N.A.	N.A.	M	
29.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	R	
30.	Turbine Trip Set Point	N.A.	R	R	Stop valve closure or low EH fluid pressure
31.	Seismic Instrumentation	M	SA	M	
32.	Reactor Trip Breaker	N.A.	N.A.	M	

S - Each Shift
D - Daily
W - Weekly
NA - Not applicable
SA - Semiannually
Q - Every 90 effective full power days

M - Monthly
P - Prior to each startup if not done previous week
R - Each Refueling Shutdown
BW - Every two weeks
AP - After each startup if not done previous week

* See Specification 4.1D

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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[6]		
	Gross Activity[2]	5 days/week[6]	9.1	
	Tritium Activity	Weekly[6]	9.1	
	*Chemistry(Cl, F&O_2)	5 days/week	4	23
	*Boron Concentration	Twice/week	9.1	
	$\bar{E}$ Determination	Semiannually[3]		
2. Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly	6	
3. Boric Acid Tanks	*Boron Concentration	Twice/week	9.1	
4. Boron Injection Tank	Boron Concentration	Twice/week	6	
5. Chemical Additive Tank	NaOH Concentration	Monthly	6	
6. Spent Fuel Pit	*Boron Concentration	Monthly	9.5	
7. Secondary Coolant	Fifteen minute degassed β and γ activity(4)	Weekly	10.3	
8. Stack Gas Iodine and Particulate Samples	*I-131 and particulate radioactive releases(5)	Weekly		
9. 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.

(3) $\bar{E}$ determination will be started when the gross gamma degassed activity of radionuclides with half-lives greater than 30 minutes analysis indicates $\geq$ 10μCi/cc.

(4) If the fifteen minute degassed beta and gamma activity is 10% of that given in Specification 3.6.C, an I-131 analysis will be performed.

(5) If the activity of the samples is 10% or greater of that given in Specification 3.11.B.1. the frequency shall be increased to daily.

(6) When reactor is critical.

Basis

The leaktightness testing of all liner welds was performed during construction by welding a structural steel test channel over each weld seam and performing soap bubble and halogen leak tests.

The containment is designed for a maximum pressure of 45 psig. The containment is maintained at a subatmospheric air partial pressure which varies between 9 psia and 11 psia depending upon the cooldown capability of the Engineered Safeguards and is not expected to rise above 45 psig for any postulated loss-of-coolant accident. 23

All loss-of-coolant accident evaluations have been based on an integrated containment leakage rate not to exceed 0.1 percent of containment volume per 24 hr.

The above specification satisfies the conditions of 10 CFR 50.54(0) which states that primary reactor containments shall meet the containment leakage test requirements set forth in Appendix J.

References

FSAR Section 5.4 Design Evaluation of Containment Tests and Inspections of Containment

FSAR Sections 7.5.1 Design Bases of Engineered Safeguards Instrumentation

FSAR Section 14.5 Loss-of-Coolant Accident

10 CFR 50 Appendix J (Proposed), "Reactor Containment Leakage Testing for Water Cooled Power Reactors," as published in the Federal Register, Volume 36, No. 167, August 27, 1971.

4.9 EFFLUENT SAMPLING AND RADIATION MONITORING SYSTEM

Applicability

Applies to the periodic monitoring and recording of radioactive effluents.

Objective

To ascertain that radioactive releases are maintained as low as practicable and within the limits set forth in 10CFR20.

Specification

A. Procedures shall be developed and used, and equipment which has been installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences, shall be maintained and used to keep levels of radioactive materials in effluents released to unrestricted areas as low as practicable.

B. All effluents to be discharged to the atmosphere from the waste gas decay tanks of the Gaseous Waste Disposal System shall be sampled prior to release via the process vent. Effluent from the Liquid Waste Disposal System shall 23
be continuously monitored by the circulating water discharge tunnel monitor and shall be sampled prior to being discharged into the circulating water discharge tunnel.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, SAFETY AND OPERATION REVIEW

Specification

A. The Station Manager shall be responsible for the safe operation of the facility. The Station Manager shall report to the Manager Production Operation and Maintenance. The relationship between this Manager and other levels of company management is shown in TS Figures 6.1-1 and 6.1-2. 23

B. The station organization shall conform to the chart as shown in TS Figure 6.1-4.

1. Qualifications with regard to education and experience and the technical specialties of key supervisory personnel will meet the minimum acceptable levels described in Regulatory Guide No. 1.8 "Personnel Selection and Training," dated March 10, 1971.

   The key supervisory personnel are as follows:

   a) Manager
   b) Superintendent-Station Operations
   c) Operating Supervisor
   d) Supervisor-Electrical Maintenance
   e) Supervisor-Mechanical Maintenance
   f) Supervisor-Engineering Services
   g) Chemistry and Health Physics Supervisor
   h) Shift Supervisor

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Safety and Operating Committee and copies shall be sent to the Manager Production Operation and Maintenance and to all members of the Station and System Nuclear Safety and Operating Committees.

h. Procedures

Written administrative procedures for committee operation shall be prepared and maintained describing the method of submission, and the content of presentations to the committee, provisions for the use of subcommittees; review and approval by members of written committee evaluations and recommendations; the distributions of minutes; and, such other matters as may be appropriate.

2. System Nuclear Safety and Operating Committee

a. Membership

1. Chairman and Vice Chairman appointed by name by the Senior Vice President-Power, which may be an individual listed in Item 2.

2. Seven members of the Power Department system office staff who are experienced in utility operation and procedures:

Manager - Production Operation and Maintenance

Manager - Licensing and Quality Assurance

Manager - Power Station Engineering 23

Design Engineer - Power Station Engineering

Director - Quality Assurance 23

Supervisor - Nuclear Fuel Design and Operation

Supervisor - Nuclear Operation

3. Manager of each nuclear generating station operating on the Virginia Electric and Power Company system, or his designee. In matters or consideration of proposals pertinent to a particular station, the Manager of this station shall serve as a non-voting member of the committee. In matters pertaining to other stations, Station Managers will serve as voting members.

4. At least one qualified non-company affiliated technical consultant. Duly appointed consultant members shall have equal vote with permanent members of the Committee.

b. Qualifications

The minimum qualifications of the Company members of the System Nuclear Safety and Operating Committee will be:

an engineering graduate or equivalent with combined nuclear and conventional experience in power station design and/or operation of eight years, with at least two years involving the direction of nuclear operations or design activity.

c. Consultants

The committee shall have the authority to call technically

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qualified personnel from within the Virginia Electric and Power Company organization or from any other consultant source.

d. Quorum: Either the Chairman or Vice Chairman and two thirds of the other members shall constitute a quorum.

e. Meeting frequency: As required by the Chairman but not less than quarterly.

f. Responsibilities

1. Review proposed changes to the operating license including Technical Specifications.

2. Review minutes of meeting of the Station Nuclear Safety and Operating Committee(s) to determine if matters considered by that Committee involve "unreviewed safety questions" as defined in 10 CFR 50.59. 23

3. Review matters including proposed changes or modifications by systems or equipment having safety significance referred to it by the Station Nuclear Safety and Operating Committee or by the Station Manager.

4. Conduct periodic review of station operations.

5. Review all reported instances of departure from Technical Specifications limits and report findings and recommendations to prevent recurrence to the Vice President-Power Supply and Production Operations. 23

Copies of the minutes shall be forwarded to the Senior Vice President-Power, Vice President-Power Supply and Production Operations, all members of the committee and any others that the Chairman may designate. 23

i. Procedures

Written administrative procedures for committee operation shall be maintained describing the method of submission and the content of presentations to the committee; provisions for use of subcommittee evaluations and recommendations; distribution of minutes; and, such other matters as may be appropriate.

TS FIG 6.1-3

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6.2 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE IN STATION OPERATION

Specification

A. Any abnormal occurrence shall be reported immediately to and promptly reviewed by the Chairman-Station Nuclear Safety and Operating Committee or his designee, Manager-Production Operations and Maintenance and the Chairman-System Nuclear Safety and Operating Committee. Prompt corrective action shall be taken to correct the anomaly.

B. A qualified individual shall prepare a written report for each abnormal occurrence for submittal to the NRC. This report shall include an evaluation of the cause of the occurrence and also recommendations for appropriate action to prevent or reduce the probability of a recurrence. 23

C. Copies of all such reports shall be submitted to the Superintendent-Station Operations, the Station Manager, who also serves as the Chairman-Station Nuclear Safety and Operating Committee, Manager-Production Operations and Maintenance, and to the Chairman-System Nuclear Safety and Operating Committee for review and approval of any recommendation.

D. A duly authorized executive of the Company, or designee, shall report the circumstances of any abnormal occurrence to the NRC as specified in Section 6.6.B.1 of these Specifications. 23

D. All procedures described in A and B above shall be followed.

E. Temporary changes to procedures described in A and B above which do not change the intent of the original procedure may be made, provided such changes are approved prior to implementation by the person designated below based on the type of procedure to be changed:

1.	Administrative	Station Manager
2.	Abnormal	Shift Supervisor
3.	Annunciator	Shift Supervisor
4.	Chemistry and Health Physics	Health Physist, Chemist
5.	Emergency	Shift Supervisor
6.	Electrical Maintenance	Electrical Foreman
7.	Mechanical Maintenance	Mechanical Foreman
8.	Operating	Shift Supervisor
9.	Periodic Test	Cognizant Supervisor 23
10.	Start-up Test	Supervisor-Engineering Services
11.	Special Test	Supervisor-Engineering Services
12.	Quality Control	Quality Control Engineer

Such changes will be documented and subsequently reviewed by the Station Nuclear Safety and Operating Committee and approved by the Station manager within seven days.

F. Temporary changes to procedures described in A and B above which change the intent of the original procedure may be made, provided such changes are approved prior to implementation by the person designated below based on the type of procedure to be changed.

1. Administrative	Station Manager
2. Abnormal	Operating Supervisor or Superintendent Station Operations
3. Annunciator	Operating Supervisor or Superintendent Station Operations
4. Chemistry and Health Physics	Supervisor-Chemistry and Health Physics
5. Emergency	Operating Supervisor or Superintendent Station Operations
6. Maintenance	Supervisor-Mechanical Maintenance Supervisor-Electrical Maintenance Instrument Supervisor
7. Operating	Operating Supervisor Superintendent-Station Operations 23
8. Periodic Test	Supervisor-Engineering Services
9. Startup Test	Supervisor-Engineering Services
10. Special Test	Supervisor-Engineering Services
11. Quality Control	Quality Control Engineer

Such changes will be documented and subsequently reviewed by the Station Nuclear Safety and Operating Committee and approved by the Station Manager.

G. In cases of emergency, operations personnel shall be authorized to depart from approved procedures where necessary to prevent injury to personnel or damage to the facility. Such changes shall be documented and reviewed by the Station Nuclear Safety and Operating Committee and approved by the Station Manager.

H. Practice of site evacuation exercises shall be conducted annually, following emergency procedures and including a check of communications with off-site report groups. An annual review of the Emergency Plan will be performed. 23

I. The industrial security program which has been established for the station shall be implemented, and appropriate investigation and/or corrective action shall be taken if the provisions of the program are violated. An annual review of the program shall be performed.

TABLE TS 6.4-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES 1/	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT NOBLE GASES AND TRITIUM OXIDE 3/	23
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 6/	NP	5	
Facepiece, full 6/	NP	100	
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Air Line respirator			
Facepiece, half-mask	CF	100	
Facepiece, full	CF	1,000	
Facepiece, full 6/	D	100	7
Facepiece, full	PD	1,000	
Hood	CF	5/	
Suit	CF	5/	
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full 6/	D	100	7
Facepiece, full	PD	1,000	
Facepiece, full	R	1,000	
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-sypplying respirator		Protection factors for type and mode of operation as listed above.	

B. Non-Routine Reports

1. Abnormal Occurrence Reports - A notification shall be made within 24 hours by telephone and telegraph to the Director of the Regional Regulatory Operations Office (cc to the Director of Licensing) followed by a written report within 15 days to the Director of the Regional Regulatory Operations Office in the event of an abnormal occurrence. The written abnormal occurrence report should follow the format presented in Regulatory Guide 1.16. 23

2. Reporting of Unusual Safety Related Events

A written report shall be forwarded within 30 days to the Director of the Regional Regulatory Operations Office in the event of an unusual safety related event.

Unusual safety related events are defined in Section 1 of these Technical Specifications.

3. Radioactive Effluents

If the experienced rate of release of radioactive materials in liquid and gaseous wastes, when averaged over any 48 hour period is such that the concentrations of radionuclides at the site boundary exceed 4% of the limits in Specification 3.11-A.1 or 3.11-B-1, a written report shall be submitted to the Deputy Director for Reactor Projects, Directorate of Licensing and to the Director,Region II Directorate of Regulatory Operations within 30 days following the release. This report shall identify the cause(s) of the activity release and describe the proposed program of action to reduce such release(s). 23

C. Inservice Inspection Reports

Notification shall be made to the Director, Region II Directorate of Regulatory Operations at least 60 days before scheduled refueling periods. This notification shall include the details of the inservice inspection planned for such period, including a detailed schedule identifying specific welds to be inspected and identifying the specific techniques to be employed.

Within 90 days after completion of inservice inspection activities, a written report shall be forwarded to the Deputy Director for Reactor Projects, Directorate of Licensing, and to the Director, Region II, Directorate of Regulatory Operations covering the scope of the examinations conducted and the results thereof. In addition, a preliminary report of the findings of the inservice inspection activities shall be filed with the Deputy Director for Reactor Projects, Directorate of Licensing, and the Director, Region II Directorate of Regulatory Operation before the facility is returned to power.

D. Special Reports

a. Startup Report - A summary report of unit startup and power escalation testing and the evaluation of the results from these test programs shall be submitted when a unit is initially placed

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 8 TO LICENSES NO. DPR-32 AND DPR-37

CHANGE NO. 23 TO TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC & POWER COMPANY

SURRY POWER STATION UNITS 1 & 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By letters dated September 6, 1974, and September 24, 1974, Virginia Electric & Power Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating Licenses No. DPR-32 and DPR-37 for the Surry Power Station Units 1 and 2. The purpose of the requests is to revise the Surry 1 and 2 Technical Specifications in order to implement 20 miscellaneous changes.

Discussion

The licensee in the above two letters requested 21 total changes but subsequently withdrew the change request for item (18) submitted in the September 6, 1974 letter.

The remaining 20 change requests will be discussed in three general groupings, seven in the Administrative Controls section of the TS, another six to correct errors of various types, and the remaining seven revisions involve minor changes to bring the TS in line with the "as-built" plant, to clarify the specifications or to make the specifications consistent with the FSAR.

The Administrative Controls group of seven TS change requests identified by date of letter and item designation are as follows:

9/06 - (12) Request proposes to eliminate a typographical error and relate TS 6.1.C.2.f.2. to 10 CFR 50.59.

9/06 - (13) The request regarding TS 6.2.A. B&D proposes to: a. speed corrective action on events requiring an abnormal occurrence (AO) report, b. to assign improved VEPCO expertise to AO report preparation, and c. to provide flexibility to VEPCO in signatory authority in regard to AO reports.

9/06 - (14) VEPCO desires to implement the requirements of American National Standards Institute (ANSI) Standard 18.7 into the Surry Unit 1 and 2 Operating Procedures by revising TS 6.4.

9/06 - (15) Request proposes to make TS Table 6.4-1 consistent with TS 6.4.B.4 in regard to noble gases and tritium oxide sample tests.

9/06 - (16) Licensee desires relief from present TS 6.6.B.1 requirement to submit a written abnormal occurrence report within ten days of event.

9/06 - (17) Request proposes to correct erratum on TS page 6.6-11.

9/24 - (3) VEPCO desires to delete Figure 6.1-3, Licensing and Quality Assurance Organization Chart, from TS.

The following group of 6 TS change requests relate to corrections of errors of various types.

9/06 - (1) Request proposes to make a change on p. TS-ii, Table of Contents, to make entry agree with TS page 4.8-1.

9/06 - (2) Request proposes to change a figure number on TS page 1.0-1.

9/06 - (3) Request proposes to change a figure number on TS page 1.0-2.

9/06 - (4) Request proposes to correct a spelling error on TS page 3.1-21.

9/06 - (5) Request proposes to correct a typographical error and a spelling error on TS page 3.6-4.

9/06 - (7) Request proposes to unscramble three improperly given references to tables on TS page 4.1-1.

The following group of 7 TS change requests relate to clarification of the specification as follows:

9/06 - (6) VEPCO desires to include in TS 3.17, in regard to the calculation of boron concentration of an isolated reactor coolant system loop, the analytical error inherent in the calculation.

9/06 - (8) VEPCO desires to clarify in TS 4.1 the means used to determine solution level in the Boron Injection Tank.

9/06 - (9) Change requests a., b., and c. relate to reactor coolant chemistry sampling requirements in Table 4.1-2B; the licensee desires to relate frequency of tests to reactor criticality. Change request d. is to correct a typographical error in Table 4.1-2B.

9/06 - (10) Request proposes to clarify a difference between the Basis of TS 4.4 and FSAR Section 5.4.

9/06 - (11) Request proposes to clarify language of TS 4.9.B to emphasize means used to sample liquid waste effluent prior to discharge.

9/24 - (1) Request proposes to clarify a conflict between TS 3.3.A.5 and TS 3.3.B.2 in regard to charging pump operation.

9/24 - (2) VEPCO desires to revise the frequency of instrument checks and calibrations of the Nuclear Power Range Monitors in Table 4.1-1 as related to plant operating experience.

Evaluation

Our evaluation of the 20 proposed changes, by groups, is as follows:

This first group of seven TS change requests is concerned with the Administrative Controls Section of the TS.

9/06 - (12) TS 6.1.C.2.f.2 contained, when originally issued, a typographical error in that the word "committee(s)" should have been singular. In addition, VEPCO in this change related the review of the minutes of the Station Nuclear Safety and Operating Committee, when involved in "unreviewed safety questions" to the requirements of 10 CFR 50.59. We conclude that this change will result in Committee reviews consistent with the requirements of 10 CFR 50.59 and is acceptable.

9/06 - (13) a. This change to TS 6.2.A provides for prompt corrective action in the event of an abnormal occurrence, a requirement that did not previously exist in the TS.
b. This change to TS 6.2.B transfers the responsibility for preparing the written AO report to NRC from the shift and operating supervisors, who may not be qualified to evaluate the cause of the occurrence nor to make corrective recommendations, to a qualified individual.

c. The present TS 6.2D restricts the preparation of AO reports to NRC to the VEPCO Senior Vice President-Power only. The change permits a duly authorized executive of VEPCO, or designee, to make the report. We conclude that the above three changes in TS 6.2 will speed corrective action in the event of an abnormal occurrence, implement assignment of more technically capable personnel to determine cause of event and to propose corrective action, and allow the licensee more signatory authority in issuing AO reports which should speed up issuance of reports to NRC, thus the changes are acceptable.

9/06 - (14) The licensee has expanded TS 6.4.E, F, and G and added TS 6.4.H and I in order to improve the unit operating procedures by implementing the requirements of ANSI Standard 18.7 into TS 6.4; in addition, annual review of the Emergency Plan and the industrial security program is specified. We conclude that these additions enhance the health and safety of the public by incorporating the ANSI Standard 18.7 and through annual review of the Emergency Plan and industrial security program, thus the changes are acceptable.

9/06 - (15) The licensee revised the heading in Table TS 6.4-1, right hand column, to exclude "noble gases" as well as tritium oxide in order to make the table consistent with TS 6.4.B.4. We conclude that this change is necessary in order to make Table TS 6.4-1 consistent with TS 6.4.B.4 which includes the Table by reference.

9/06 - (16) The licensee requested relief from the requirement to file abnormal occurrence reports within 10 days of the event, in this change to TS 6.6.B.1. The licensee desired a 30 day delay period. In discussion with the licensee we pointed out that Regulatory Guide 1.16-Revision 3 calls for a report within 15 days of the event. The licensee agreed to the 15 day reporting limit. We conclude that a 15 day AO reporting limit in TS 6.6.B.1 is acceptable as it meets the position of RG 1.16-Revision 3.

9/06 - (17) The last word of TS 6.6.B.3, "release(s)," was inadvertently omitted when the TS was originally issued. This change adds the required word and thus is acceptable.

9/24 - (3) The licensee requests the deletion of Figure 6.1-3, Licensing and Quality Assurance Organization Chart, from the TS. The deletion of the chart is acceptable as it is not referenced in the TS and thus does not constitute an actual TS requirement.

The above group of seven changes in the Administrative Controls section of the Technical Specifications does not result from any unreviewed safety questions and thus do not involve any significant hazards considerations.

The following group of six TS change requests relate to the correction of errors of various types.

9/06 - (1) The licensee requests that the Table of Contents of the TS, on page TS-ii, be made to agree with TS page 4.8-1. The change is acceptable.

9/06 - (2) This change corrects an error in the original TS on page TS 1.0-1 where Figure 3.12-7 was improperly given as Figure 3.12-2 and thus is acceptable.

9/06 - (3) This change corrects an error in the original TS on page TS 1.0-2 where Figure 3.12-7 was improperly given as Figure 3.12-2 and thus is acceptable.

9/06 - (4) This change corrects an error in spelling the word "consistent" on page TS 3.1-21 of the original TS and thus is acceptable.

9/06 - (5) This change removes one of the words "reactor" from the sentence starting on page TS 3.6-3 and ending on page TS 3.6-4 reading originally as follows, ". . . two or three reactor reactor coolant loop operation." In addition, the misspelling of the word "dose" on page TS 3.6-4 is corrected. These corrections are acceptable.

9/06 - (7) This change reflects the fact that "Table 4.1-2" as given in TS 4.1 is in reality two tables, 4.1-2A and 4.1-2B; in addition, a typographical error wherein TS 4.1.C referenced "Table 4.2-2B" is corrected to read Table 4.1-2B. Finally, TS 4.1.C is corrected to reflect the fact that Table 4.1-3 was split on January 17, 1973, into two tables 4.1-3A and 4.1-3B in order to reflect differences between Surry Unit 1 and Unit 2 in regard to the flushing of sections of stainless steel pipe.

These changes serve to properly reference Tables 4.1-2A, 4.1-2B, 4.1-3A and 4.1-3B in TS 4.1, and to correct a typographical error and are acceptable.

The above group of six changes in the Technical Specifications serve to correct various errors and thus does not involve any significant hazards considerations.

The following group of seven change requests relates to clarifications of the TS.

9/06 - (6) The licensee desires to include the inherent analytical error in calculating boron concentrations in an isolated reactor coolant loop by inserting the phrase "($\pm$ 2% analytical error)" in TS 3.17. This TS applies to the operation of the loop stop valves. The valves permit any one of the three loops to be isolated from the reactor vessel. There is one loop stop valve in each hot leg and each cold leg. Before a loop is reopened the present technical specifications require that the boron concentration in the idle loop be brought to the same concentration as the remainder of the reactor coolant system, in order to avoid any sudden increase or decrease in core reactivity. However, no provision is made for unavoidable analytical error. The proposed change specifies an approporate limit for this error. In balancing the concentrations there is an inherent error in the analytical method which at the extreme could be a 4% variation (e.g., +2% in the loop, -2% in the RCS). Our review of the FSAR Section 14.2.6 indicates that even if the isolated loop had 0 parts per million (ppm) of boron and the remainder of the system had 1500 ppm, the short term insertion of the isolated loop coolant would have a negligible effect on core reactivity, thus a variation in concentrations of the extreme difference of 4% would have a negligible effect on core reactivity. Therefore, the change is acceptable.

9/06 - (8) The licensee desires to change TS Table 4.1-1, line item No. 16, in order to delete the requirement to calibrate the Boron Injection Tank level detector. This change is acceptable as the tank does not have a level detector but indicates through continuous orifice flow in a pipe line at the top of the tank. This flow indicates that the tank is full. The inclusion of a requirement to calibrate this detector in the original TS was an oversight.

9/06 - (9) The licensee desires to change TS Table 4.1-2B, line item No. 1, in order to limit the requirement to perform radio-chemical analysis, gross activity and tritium activity of the reactor coolant, only to periods when the reactor is critical. The TS presently require these measurements to be made on a calendar basis, irrespective of the operational mode of the facility. Footnote (6) is added to the Table to state that the above three mentioned samples are taken only when the reactor is critical.

We conclude the change is acceptable based on the following:

The radio-chemical analysis is only indicative of the true level of corrosion product ionic concentration when the reactor is at power. Therefore such an analysis does not serve a useful purpose at times when the reactor is not critical. Also when the reactor is not critical, the gross activity will predictably decrease. Also, since tritium production is practically negligible at this time, there is no need for a tritium activity test during a non-critical period.

The final listed test in line item 1 is an "$\bar{E}$" not "E" determination thus this is an acceptable correction of a typographical error in Table 4.1-2B.

9/06 - (10) The licensee desires to change the first sentence of the second paragraph of the Basis of TS 4.4 to read, "The containment is designed for a maximum pressure of 45 psig." The change is acceptable as it makes the Basis of TS 4.4 agree with FSAR Section 5.4.

9/06 - (11) The licensee desires to change the second sentence of TS 4.9.B to read, "Effluent from the Liquid Waste Disposal System shall be continuously monitored by the circulating water discharge tunnel monitor and shall be sampled prior to being discharged into the circulating water discharge tunnel." This change is acceptable as it serves to clarify TS 4.9.B and emphasize the point that effluent is sampled prior to discharge into the discharge tunnel to reflect "as-built" status.

9/24 - (1) The licensee desires to delete the second sentence of TS 3.3.B.2 which relates to the testing of an operable charging pump. There are three charging pumps any one of which is sufficient for either normal plant operation or to serve as a high pressure coolant injection pump in the unlikely event of a loss-of-coolant accident. TS 3.3.B.2 states that two pumps may be out of service, which is in accord with the FSAR Safety Analysis of Section 14.5.2, provided one of the two such pumps is restored to operable status in 24 hours. This provision of TS 3.3.B.2 is not being modified. The sentence the licensee wants deleted states, "If one pump unit is out of service, the standby pump shall be tested before initiating maintenance and once every 8 hours to assure operability." The determination of the operable status of a pump would have to be made by a pump test, thus such a test is inherent in the first sentence of TS 3.3.B.2 and the second sentence imposes an unnecessary restriction and is inconsistent with the authorization contained in the first sentence. We conclude that the deletion of the second sentence of TS 3.3.B.2 would remove an unnecessary restriction on charging pump operation and permits pump operation within the envelope of the FSAR safety analysis; thus this change is acceptable.

9/24 - (2) The licensee desires in TS Table 4.1-1 to change the calibration requirements of the nuclear power range monitors from monthly to "every 90 effective full power days," by use of the moveable incore detector system. The monitors will continue to be calibrated daily, against a heat balance standard. The licensee proposes to not only check the system each shift but to add a monthly check by means of the moveable incore detector system. This additional monthly check of the monitors by use of the moveable incore detector system both compensates for the extension of time for the monitor calibration from monthly to every 90 EFPD and provides that the need for any additional calibration will be determined on a monthly basis. We conclude based on the licensee's experience at Surry, and verified by Westinghouse at other PWRs, that the change in monitor channel response time due to burnup is small thus the test and calibration frequency changes are acceptable.

The above group of seven changes does not involve any changes in limiting conditions for operations, does not involve or result from an unreviewed safety question nor revises any safety limit settings. We conclude the changes do not involve any significant hazards considerations.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change 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 change 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: JUL 2 2 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC & POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 8 to Facility Operating Licenses No. DPR-32 and DPR-37 issued to Virginia Electric & Power Company which revised Technical Specifications for operation of the Surry Power Station, Units 1 and 2, located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments revise the provisions in the Technical Specifications relating to 20 miscellaneous items. Seven of the revisions involve the Administrative Controls section of the Technical Specifications, another six correct errors of various types, and the remaining seven revisions involve minor changes to bring the Technical Specifications in line with the "as-built" plant, to clarify the specifications or to make the specifications consistent with the Final Safety Analysis Report (FSAR).

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

For further details with respect to this action, see (1) the applications for amendments dated September 6, 1974 and September 24, 1974, (2) Amendments No. 8 to Licenses No. DPR-32 and DPR-37, with Change No. 23 , 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, NW, Washington, D.C. and at the Swem Library, College of William & 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 Reactor Licensing.

Dated at Bethesda, Maryland, this 22nd day of July 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

V. L. Rooney

Vernon L. Rooney, Acting Branch Chief
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