July 8, 1993

Docket Nos. 50-280
and 50-281

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Mr. W. L. Stewart
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Stewart:

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: RECIRCULATION MODE TRANSFER FUNCTION (TAC NOS. M84310 AND M84311)

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-32 and Amendment No. 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 (TS) in response to your application transmitted by letter dated August 7, 1992.

These amendments establish operating requirements for the recirculation mode transfer function (RMT) and include the following: 1) Limiting Conditions for Operation, Action Statements, and Surveillance Requirements for the RMT function, 2) deletion of refueling water storage tank maximum volume requirements and clarification of refueling water storage tank water temperature requirements, and 3) administrative changes to provide consistency throughout.

A copy of the Safety Evaluation and the Notice of Issuance are enclosed.

Sincerely,
(Original Signed By)
Bart C. Buckley, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 180 to DPR-32
2. Amendment No. 180 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

*Previously Concurred
Document Name - SU84310.AMD

OFC	:LA:PDII-2*	:PM:PDII-2	:D:PDII-2	: HICB*	:OTSB*	:OGC
	EH	BCB				
NAME	:E. Tana	:B. Buckley	:H. Berkow	:J. Wermiel	:C. Grimes	: EHolen
	6/22/93					
DATE	:~~05/27/93~~	: 6/22/93	: 6/22/93	:06/09/93	:06/21/93	: 6/25/93

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Surry Power Station

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DATED: July 8, 1993

AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-32 - SURRY UNIT 1
AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-37 - SURRY UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
E. Tana
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OGC
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, MNBB-7103
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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. 180
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 August 7, 1992, 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. 180 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 8, 1993

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. 180
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 August 7, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

    (B) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 180 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 8, 1993

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
1.0-1 thru 1.0-10	1.0-1 thru 1.0-7
3.3-1 thru 3.3-9	3.3-1 thru 3.3-7
3.4-1 thru 3.4-6	3.4-1 thru 3.4-4
3.7-1 thru 3.7-22	3.7-1 thru 3.7-29
4.1-7	4.1-7
4.1-9d	4.1-9d

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

When the reactor is subcritical by at least 5% Δk/k and T_{avg} is ≤140°F and fuel is scheduled to be moved to or from the reactor core.

2. COLD SHUTDOWN

When the reactor is subcritical by at least 1% Δk/k and T_{avg} is ≤200°F.

3. INTERMEDIATE SHUTDOWN

When the reactor is subcritical by at least 1.77% Δk/k and 200°F < T_{avg} < 547°F.

4. HOT SHUTDOWN

When the reactor is subcritical by at least 1.77% Δk/k and T_{avg} is ≥ 547°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

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

D. OPERABLE

A system, subsystem, train, component, or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s). 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.

E. PROTECTIVE INSTRUMENTATION LOGIC

1. ANALOG CHANNEL

An arrangement of components and modules as required to generate a single protective action digital signal when required by a unit condition. An analog channel loses its identity when single action signals are combined.

2. AUTOMATIC ACTUATION LOGIC

A group of matrixed relay contacts which operate in response to the digital output signals from the analog channels to generate a protective action signal.

F. INSTRUMENTATION SURVEILLANCE

1. CHANNEL CHECK

The qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation on channels measuring the same parameter.

2. CHANNEL FUNCTIONAL TEST

Injection of a simulated signal into an analog channel as close to the sensor as practicable or makeup of the logic combinations in a logic channel to verify that it is operable, including alarm and/or trip initiating action.

3. CHANNEL CALIBRATION

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarm, or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

G. CONTAINMENT INTEGRITY

Containment integrity shall exist when:

a. The penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by at least one closed manual valve, blind flange, or deactivated automatic valve secured in its closed position except as provided in Specification 3.8.C. Non-automatic or deactivated automatic containment isolation valves may be opened intermittently for operational activities provided that the valves are under administrative control and are capable of being closed immediately, if required.

b. The equipment access hatch is closed and sealed.

c. Each airlock is OPERABLE except as provided in Specification 3.8.B.

d. The containment leakage rates are within the limits of Specification 4.4.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

H. <u>REPORTABLE EVENT</u>

A reportable event shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

I. <u>QUADRANT POWER TILT</u>

The quadrant power tilt is defined as the ratio of the maximum upper excore detector current to the average of the upper excore detector currents or the ratio of the maximum lower excore detector current to the average of the lower excore detector currents whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

J. <u>LOW POWER PHYSICS TESTS</u>

Low power physics tests conducted below 5% of rated power which measure fundamental characteristics of the core and related instrumentation.

K. FIRE SUPPRESSION WATER SYSTEM

A fire suppression water system shall consist of: a water source(s), gravity tank(s) or pump(s), and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

L. OFFSITE DOSE CALCULATION MANUAL (ODCM)

The Offsite Dose Calculation Manual (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specifications 6.6.B.2 and 6.6.B.3.

M. DOSE EQUIVALENT I-131

The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

N. GASEOUS RADWASTE TREATMENT SYSTEM

A gaseous radwaste treatment system is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

O. PROCESS CONTROL PROGRAM (PCP)

The process control program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, and other requirements governing the disposal of the waste.

P. PURGE - PURGING

Purge or purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

Q. VENTILATION EXHAUST TREATMENT SYSTEM

A ventilation exhaust treatment system is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents. Treatment includes passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.

R. VENTING

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

S. SITE BOUNDARY

The site boundary shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

T. UNRESTRICTED AREA

An unrestricted area shall be any area at or beyond the site boundary where access is not controlled by the licensee for purpose of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or for industrial, commercial, institutional, or recreational purposes.

U. MEMBER(S) OF THE PUBLIC

Member(s) of the public shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

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 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 at least 387,100 gallons of borated water at a maximum temperature of 45°F. The boron concentration shall be at least 2300 ppm but not greater than 2500 ppm.

2. Each accumulator system is pressurized to at least 600 psia and contains a minimum of 975 ft^3 and a maximum of 1025 ft^3 of borated water with a boron concentration of at least 2250 ppm.

3. Two channels of heat tracing shall be OPERABLE 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 AC power shall be removed from the following motor-operated valves with the valves in the open position:

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

9. During POWER OPERATION, the AC power shall be removed from the following motor-operated valves with the valves 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 shall be blocked open by de-energizing the valves motor operators 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 specified by Technical Specification 4.11.A.4.d.

B. The requirements of Specification 3.3.A may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of Specification 3.3.A within the time period specified, the reactor shall be placed in HOT SHUTDOWN within the next 6 hours. If the requirements of Specification 3.3.A are not satisfied within an additional 48 hours the reactor shall be placed in COLD SHUTDOWN within the next 30 hours.

1. One accumulator may be isolated for a period not to exceed 4 hours.

2. Two charging pumps per unit may be inoperable, provided immediate attention is directed to making repairs and one of the inoperable pumps is restored to OPERABLE status within 24 hours.

3. One low head safety injection subsystem per unit may be inoperable provided immediate attention is directed to making repairs and the subsystem is restored to OPERABLE status within 24 hours.

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

5. One charging pump component cooling water pump or one charging pump service water pump may be inoperable provided the pump is restored to OPERABLE status within 24 hours.

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

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

8. Power may be restored to any valve referenced in Specification 3.3.A.10 for the purpose of valve testing or maintenance, provided that no more than one valve has power restored and the testing and maintenance is completed and power removed within 4 hours.

9. The total uncollected system leakage for valves, flanges, and pumps located outside containment can exceed the limit stated in Specification 4.11.A.4.d provided immediate attention is directed to making repairs and uncollected system leakage is returned to within limits within 7 days.

10. Refueling water storage tank volume, temperature, and boron concentration may be outside the limits of Specification 3.3.A.1 provided they are restored to within their respective limits within one hour.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant. With this mode of startup the Safety Injection System is required to be OPERABLE as specified. During LOW POWER PHYSICS TESTS there is a negligible amount of energy stored in the system. Therefore, an accident comparable in severity to the Design Basis Accident is not possible, and the full capacity of the Safety Injection System would not be necessary.

The OPERABLE status of the various systems and components is to be demonstrated by periodic tests, detailed in TS Section 4.11. A large fraction of these tests are performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. A single component being inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. In some cases, additional components (i.e., charging pumps) are installed to allow a component to be inoperable without affecting system redundancy.

If the inoperable component is not repaired within the specified allowable time period, or a second component in the same or related system is found to be inoperable, the reactor will initially be placed in HOT SHUTDOWN to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. After 48 hours in HOT SHUTDOWN, if the malfunction(s) is not corrected the reactor will be placed in COLD SHUTDOWN following normal shutdown and cooldown procedures.

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

Assuming the reactor has been operating at full RATED POWER for at least 100 days, the magnitude of the decay heat production decreases as follows after a unit trip from full RATED POWER.

Time After Shutdown	Decay Heat, % of RATED POWER
1 min.	3.7
30 min.	1.6
1 hour	1.3
8 hours	0.75
48 hours	0.48

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

Failure to complete repairs within 48 hours of going to HOT SHUTDOWN is considered indicative of unforeseen problems (i.e., possibly the need of major maintenance). In such a case, the reactor is placed in COLD SHUTDOWN.

The accumulators are able to accept leakage from the Reactor Coolant System without any effect on their operability. Allowable inleakage is based on the volume of water than can be added to the initial amount without exceeding the volume given in Specification 3.3.A.2. The maximum acceptable inleakage is 50 cubic feet per tank.

The accumulators (one for each loop) discharge into the cold leg of the reactor coolant piping when Reactor Coolant System pressure decreases below accumulator pressure, thus assuring rapid core cooling for large breaks. The line from each accumulator is provided with a motor-operated 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. However, 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 accumulator injection is initiated when this pressure drops to 600 psia. De-energizing the motor operator when the pressure exceeds 1000 psig allows sufficient time during normal startup operation to perform the actions required to de-energize the valve. This procedure will assure that there is an OPERABLE flow path from each accumulator to the Reactor Coolant System during POWER OPERATION and that safety injection can be accomplished.

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

Total system uncollected leakage is controlled to limit offsite doses resulting from system leakage after a loss-of-coolant accident.

* For Unit 2 Cycle 12, Reactor Coolant System nominal operating pressure may be reduced to 2135 psig.

3.4 SPRAY SYSTEMS

Applicability

Applies to the operational status of the Spray Systems.

Objective

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

Specification

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

1. Two Containment Spray Subsystems, including containment spray pumps, piping, and valves shall be OPERABLE.

2. Four Recirculation Spray Subsystems, including recirculation spray pumps, coolers, piping, and valves shall be OPERABLE.

3. The refueling water storage tank shall contain at least 387,100 gallons of borated water at a maximum temperature of 45°F. The boron concentration shall be at least 2300 ppm but not greater than 2500 ppm.

4. The refueling water chemical addition tank shall contain at least 4,200 gallons of solution with a sodium hydroxide concentration of at least 17 percent by weight but not greater than 18 percent by weight.

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

6. The total uncollected system leakage from valves, flanges, and pumps located outside containment shall not exceed the limit specified by Specification 4.5.B.4.

B. During POWER OPERATION the requirements of Specification 3.4.A may be modified to allow a subsystem or the following components to be inoperable. If the components are not restored to meet the requirements of Specification 3.4.A within the time period specified below, the reactor shall be placed in HOT SHUTDOWN within the next 6 hours. If the requirements of Specification 3.4.A are not satisfied within an additional 48 hours the reactor shall be placed in COLD SHUTDOWN within the next 30 hours.

1. One Containment Spray Subsystem may be inoperable, provided immediate attention is directed to making repairs and the subsystem can be restored to OPERABLE status within 24 hours.

2. One outside Recirculation Spray Subsystem may be inoperable, provided immediate attention is directed to making repairs and the subsystem can be restored to OPERABLE status within 24 hours.

3. One inside Recirculation Spray Subsystem may be inoperable, provided immediate attention is directed to making repairs and the subsystem can be restored to OPERABLE status within 72 hours.

4. The total uncollected system leakage from valves, flanges, and pumps located outside containment can exceed the limit stated in Specification 4.5.B.4, provided immediate attention is directed to making repairs and uncollected system leakage is returned to within limits within 7 days.

5. Refueling Water Storage Tank volume, temperature, and boron concentration may be outside the limits of Specification 3.4.A.3 provided they are restored to within their respective limits within one hour.

Basis

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

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

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

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

In addition to supplying water to the Containment Spray System, the refueling water storage tank is also a source of water for safety injection following an accident. This water is borated to a concentration which assures reactor shutdown by approximately 5 percent Δk/k when all control rod assemblies are inserted and when the reactor is cooled down for refueling.

Total system uncollected leakage is controlled to limit offsite doses resulting from system leakage after a loss-of-coolant accident.

<u>References</u>

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

3.7 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to reactor and safety features instrumentation systems.

Objectives

To ensure the automatic initiation of the Reactor Protection System and the Engineered Safety Features in the event that a principal process variable limit is exceeded, and to define the limiting conditions for operation of the plant instrumentation and safety circuits necessary to ensure reactor and plant safety.

Specification

A. During on-line testing or in the event of a subsystem instrumentation channel failure, plant operation at RATED POWER shall be permitted to continue in accordance with Tables 3.7-1 through 3.7-3.

B. The Reactor Protection System instrumentation channels and interlocks shall be OPERABLE as specified in Table 3.7-1.

C. The Engineered Safeguards Actions and Isolation Function Instrumentation channels and interlocks shall be OPERABLE as specified in Tables 3.7-2 and 3.7-3, respectively.

D. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.7-4.

E. The explosive gas monitoring instrumentation channel shown in Table 3.7-5(a) shall be OPERABLE with its alarm setpoint set to ensure that the limits of Specification 3.11.A.1 are not exceeded.

   1. With an explosive gas monitoring instrumentation channel alarm setpoint less conservative than required by the above specification, declare the channel inoperable and take the action shown in Table 3.7-5(a).

2. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the action shown in Table 3.7-5(a). Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission (Region II) to explain why this inoperability was not corrected in a timely manner.

F. The accident monitoring instrumentation listed in Table 3.7-6 shall be OPERABLE in accordance with the following:

1. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.7-6, items 1 through 9, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum OPERABLE Channels requirement of Table 3.7-6, items 1 through 9, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

G. The containment hydrogen analyzers and associated support equipment shall be OPERABLE in accordance with the following:

1. Two independent containment hydrogen analyzers shall be OPERABLE during REACTOR CRITICAL or POWER OPERATION.

a. With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 6 hours.

b. With both hydrogen analyzers inoperable, restore at least one analyzer to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 6 hours.

NOTE: Operability of the hydrogen analyzers includes proper operation of the respective Heat Tracing System.

Basis

Instrument Operating Conditions

During plant operations, the complete instrumentation system will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines the limiting conditions for operation necessary to preserve the effectiveness of the Reactor Protection System when any one or more of the channels is out of service.

Almost all Reactor Protection System channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode (e.g., a two-out-of-three circuit becomes a one-out-of-two circuit). The Nuclear Instrumentation System (NIS) channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the dropped-rod protection from NIS, for the channel being tested, (b) placing the $\Delta T/T_{avg}$ protection channel set that is being fed from the NIS channel in the trip mode, and (c) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features.(1)

Safety Injection System Actuation

Protection against a loss-of-coolant or steam line break accident is provided by automatic actuation of the Safety Injection System (SIS) which provides emergency cooling and reduction of reactivity.

The loss-of-coolant accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The engineered safeguards instrumentation has been designed to sense these effects of the loss-of-coolant accident by detecting low pressurizer pressure to generator signals actuating the SIS active phase. The SIS active phase is also actuated by a high containment pressure signal brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between the steam header and steam generator line or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason, protection against a steam line break accident is also provided by low pressurizer pressure actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features also initiate containment spray upon sensing a high-high containment pressure signal. The containment spray acts to reduce containment pressure in the event of a loss-of-coolant or steam line break accident inside the containment. The containment spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure than the SIS. Since spurious actuation of containment spray is to be avoided, it is initiated only on coincidence of high-high containment pressure sensed by 3 out of the 4 containment pressure signals.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing the steam line trip valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high-high containment pressure or high steam line flow with coincident low steam line pressure or low reactor coolant average temperature. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the SIS in order to prevent excessive cooldown of the Reactor Coolant System. This mitigates the effects of an accident such as a steam line break which in itself causes excessive coolant temperature cooldown. Feedwater line isolation also reduces the consequences of a steam line break inside the containment by stopping the entry of feedwater.

Auxiliary Feedwater System Actuation

The automatic initiation of auxiliary feedwater flow to the steam generators by instruments identified in Table 3.7-2 ensures that the Reactor Coolant System decay heat can be removed following loss of main feedwater flow. This is consistent with the requirements of the "TMI-2 Lessons Learned Task Force Status Report," NUREG-0578, item 2.1.7.b.

Setting Limits

1. The high containment pressure limit is set at about 10% of design containment pressure. Initiation of safety injection protects against loss of coolant(2) or steam line break(3) accidents as discussed in the safety analysis.

2. The high-high containment pressure limit is set at about 23% of design containment pressure. Initiation of containment spray and steam line isolation protects against large loss-of-coolant(2) or steam line break accidents(3) as discussed in the safety analysis.

3. The pressurizer low pressure setpoint for safety injection actuation is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis.(2)

4. The steam line high differential pressure limit is set well below the differential pressure expected in the event of a large steam line break accident as shown in the safety analysis.(3)

5. The high steam line flow differential pressure setpoint is constant at 40% full flow between no load and 20% load and increasing linearly to 110% of full flow at full load in order to protect against large steam line break accidents. The coincident low T_{avg} setting limit for SIS and steam line isolation initiation is set below its HOT SHUTDOWN value. The coincident

steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break.(3)

Accident Monitoring Instrumentation

The operability of the accident monitoring instrumentation in Table 3.7-6 ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. On the pressurizer PORVs, the pertinent channels consist of redundant limit switch indication. The pressurizer safety valves utilize an acoustic monitor channel and a downstream high temperature indication channel. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations." Potential accident effluent release paths are equipped with radiation monitors to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. The effluent release paths monitored are the process vent stack, ventilation vent stack, main steam safety valve and atmospheric dump valve discharge and the AFW pump turbine exhaust. These monitors meet the requirements of NUREG 0737.

Instrumentation is provided for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the Waste Gas Holdup System. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

Containment Hydrogen Analyzers

Indication of hydrogen concentration in the containment atmosphere is provided in the control room over the range of zero to ten percent hydrogen concentration.

These redundant, qualified hydrogen analyzers are shared by Units 1 and 2 with instrumentation to indicate and record the hydrogen concentration.

A transfer switch is provided for Unit 1 to use both analyzers or for Unit 2 to use both analyzers. In addition, each unit's hydrogen analyzer has a transferable emergency power supply from Unit 1 and Unit 2. This will ensure redundancy for each unit.

Indication of Unit 1 and Unit 2 hydrogen concentration is provided on the Unit 1 Post Accident Monitoring panel and the Unit 2 Post Accident Monitoring panel, respectively. Hydrogen concentration is also recorded on qualified recorders. In addition, each hydrogen analyzer is provided with an alarm for trouble/high hydrogen content. These alarms are located in the control room.

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

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

Non-Essential Service Water Isolation System

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

<u>References</u>

(1) UFSAR - Section 7.5
(2) UFSAR - Section 14.5
(3) UFSAR - Section 14.3.2

TABLE 3.7-1

REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
1. Manual	2	2	1		1
2. Nuclear Flux Power Range	4	3	2	Low trip setting at P-10	2
3. Nuclear Flux Intermediate Range	2	2	1	P-10	3
4. Nuclear Flux Source Range				P-6	
a. Below P-6 - Note A	2	2	1		4
b. Shutdown - Note B	2	1	0		5
5. Overtemperature ΔT	3	2	2		6
6. Overpower ΔT	3	2	2		6
7. Low Pressurizer Pressure	3	2	2	P-7	7
8. Hi Pressurizer Pressure	3	2	2		7

Note A - With the reactor trip breakers closed and the control rod drive system capable of rod withdrawal.

Note B - With the reactor trip breakers open.

TABLE 3.7-1

REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
9.	Pressurizer-Hi Water Level	3	2	2	P-7	6
10.	Low Flow	3/loop	2/loop in each operating loop	2/loop in any operating loop	P-8	6
				2/loop in any 2 operating loops	P-7	
11.	Turbine Trip					
	a. Stop valve closure	4	1	4	P-7	12
	b. Low fluid oil pressure	3	2	2	P-7	6
12.	Lo-Lo Steam Generator Water Level	3/loop	2/loop in each operating loop	2/loop in any operating loops		7
13.	Underfrequency 4KV Bus	3-1/bus	2	2	P-7	6
14.	Undervoltage 4KV Bus	3-1/bus	2	2	P-7	7
15.	Safety Injection (SI) Input From ESF	2	2	1		8A
16.	Reactor Coolant Pump Breaker Position	1/breaker	1/breaker per operating loop	1	P-8	9
				2	P-7	10

TABLE 3.7-1

REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
17.	Low steam generator water level with steam/feedwater flow mismatch	2/loop-level and 2/loop-flow mismatch	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1/loop-level coincident with 1/loop-flow mismatch in same loop		7
18.	a. Reactor Trip Breakers	2	2	1		8
	b. Reactor Trip Bypass Breakers - Note C	2	1	1		
19.	Automatic Trip Logic	2	2	1		11
20.	Reactor Trip System Interlocks - Note D					
	a. Intermediate range neutron flux, P-6	2	2	1		13
	b. Low power reactor trips block, P-7					
	Power range neutron flux, P-10 and	4	3	2		13
	Turbine impulse pressure	2	2	1		13
	c. Power range neutron flux, P-8	4	3	2		13
	d. Power range neutron flux, P-10	4	3	2		13
	e. Turbine impulse pressure	2	2	1		13

Note C - With the Reactor Trip Breaker open for surveillance testing in accordance with Specification Table 4.1-1 (Item 30)
Note D - Reactor Trip System Interlocks are described in Table 4.1-A

TABLE 3.7-1 (Continued)
TABLE NOTATION

ACTION STATEMENTS

ACTION 1. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 6 hours and/or open the reactor trip breakers.

ACTION 2.A. With the number of OPERABLE channels equal to the Minimum OPERABLE Channels, POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 6 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of the redundant channel(s) per Specification 4.1.

3. Either, THERMAL POWER is restricted to $\leq$ 75% of RATED POWER and the Power Range, Neutron Flux trip setpoint is reduced to $\leq$ 85% of RATED POWER within 4 hours; or, the QUADRANT POWER TILT is monitored at least once per 12 hours.

TABLE 3.7-1 (Continued)

4. The QUADRANT POWER TILT shall be determined to be within the limit when above 75 percent of RATED POWER with one Power Range Channel inoperable by using the moveable incore detectors to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT at least once per 12 hours.

2.B. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, be in HOT SHUTDOWN within 6 hours

ACTION 3. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement and with the THERMAL POWER level:

a. Below the P-6 (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Block of Source Range Reactor Trip) setpoint, but below 10% of RATED POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED POWER.

c. Above 10% of RATED POWER, POWER OPERATION may continue.

TABLE 3.7-1 (Continued)

ACTION 4. With the number of channels OPERABLE one less than required by the Minimum OPERABLE Channels requirement and with the THERMAL POWER level:

a. Below P-6, (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. Two Source Range channels must be OPERABLE prior to increasing THERMAL POWER above the P-6 setpoint.

b. Above P-6, operation may continue.

ACTION 5. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, verify compliance with the Shutdown Margin requirements within 1 hour and at least once per 12 hours thereafter.

ACTION 6.A. With the number of OPERABLE channels equal to the Minimum OPERABLE Channels requirement, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 6 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.1.

6.B. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, be in HOT SHUTDOWN within 6 hours.

TABLE 3.7-1 (Continued)

ACTION 7. With the number of OPERABLE channels equal to the Minimum OPERABLE Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 6 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1.

ACTION 8.A. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 6 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

8.B. With one of the diverse trip features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply Action 8.A. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.7-1 (Continued)

ACTION 9.	With one channel inoperable, restore the inoperable channel to OPERABLE status within 6 hours or reduce THERMAL POWER to below the P-8 (Block of Low Reactor Coolant Pump Flow and Reactor Coolant Pump Breaker Position) setpoint within the next 2 hours. Operation below P-8 may continue pursuant to ACTION 10.
ACTION 10.	With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, operation may continue provided the inoperable channel is placed in the tripped condition within 6 hours.
ACTION 11.	With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 8 hours. In conditions of operation other than REACTOR CRITICAL or POWER OPERATIONS, with the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour. However, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.
ACTION 12.	With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.
ACTION 13.	With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

TABLE 3.7-2

ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
1. SAFETY INJECTION (SI)					
a. Manual	2	2	1		21
b. High containment pressure	4	3	3		17
c. High differential pressure between any steam line and the steam header	3/steam line	2/steam line	2/steam line on any steam line	Primary pressure less than 2000 psig, except when reactor is critical	20
d. Pressurizer low-low pressure	3	2	2	Primary pressure less than 2000 psig, except when reactor is critical	20
e. High steam flow in 2/3 steam lines coincident with low T_{avg} or low steam line pressure					
1) Steam line flow	2/steam line	1/steam line	1/steam line any two lines	Reactor coolant T_{avg} less than 543° during heatup and cooldown	20
2) T_{avg}	1/loop	1/loop any two loops	1/loop any two loops	Reactor coolant T_{avg} less than 543° during heatup and cooldown	20
3) Steam line pressure	1/line	1/line any two loops	1/line any two loops	Reactor coolant T_{avg} less than 543° during heatup and cooldown	20
f. Automatic actuation logic	2	2	1		14

TABLE 3.7-2 (Continued)

ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
2. CONTAINMENT SPRAY					
a. Manual	1 set	1 set	1 set♦		15
b. High containment pressure (Hi-Hi)	4	3	3		17
c. Automatic actuation logic	2	2	1		14
3. AUXILIARY FEEDWATER					
a. Steam generator water level low-low					
1) Start motor driven pumps	3/steam generator	2/steam generator	2/steam generator any 1 generator		20
2) Starts turbine driven pump	3/steam generator	2/steam generator	2/steam generator any 2 generators		20
b. RCP undervoltage starts turbine driven pump	3	2	2		20
c. Safety injection - start motor driven pumps	See #1 above (all SI initiating functions and requirements)				
d. Station blackout - start motor driven pumps	1/bus 2 transfer buses/unit	1/bus 2 transfer buses/unit	2		21

♦ Must actuate 2 switches simultaneously

TABLE 3.7-2 (Continued)

ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
AUXILIARY FEEDWATER (continued)					
e. Trip of main feedwater pumps - start motor driven pumps	2/MFW pump	1/MFW pump	2-1 each MFW pump		21
f. Automatic actuation logic	2	2	1		22
4. LOSS OF POWER					
a. 4.16 kv emergency bus undervoltage (loss of voltage)	3/bus	2/bus	2/bus		20
b. 4.16 kv emergency bus undervoltage (degraded voltage)	3/bus	2/bus	2/bus		20
5. NON-ESSENTIAL SERVICE WATER ISOLATION					
a. Low intake canal level	4	3	3		20
b. Automatic actuation logic	2	2	1		14
6. ENGINEERED SAFEGUARDS ACTUATION INTERLOCKS-Note A					
a. Pressurizer pressure, P-11	3	2	2		23
b. Low-low T_{avg}, P-12	3	2	2		23
c. Reactor trip, P-4	2	2	1		24
7. RECIRCULATION MODE TRANSFER					
a. RWST Level - Low	4	3	2		25
b. Automatic Actuation Logic and Actuation Relays	2	2	1		14

Note A - Engineered Safeguards Actuation Interlocks are described in Table 4 .1-A

TABLE 3.7-3

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
1. CONTAINMENT ISOLATION					
a. Phase I					
1) Safety Injection (SI)	See Item #1, Table 3.7-2 (all SI initiating functions and requirements)				
2) Automatic initiation logic	2	2	1		14
3) Manual	2	2	1		21
b. Phase 2					
1) High containment pressure	4	3	3		17
2) Automatic actuation logic	2	2	1		14
3) Manual	2	2	1		15
c. Phase 3					
1) High containment pressure (Hi-Hi setpoint)	4	3	3		17
2) Automatic actuation logic	2	2	1		14
3) Manual	1 set	1 set	1 set♦		15
2. STEAMLINE ISOLATION					
a. High steam flow in 2/3 lines coincident with 2/3 low T_{avg} or 2/3 low steam pressures	See Item #1.e Table 3.7-2 for operability requirements				

♦ Must actuate 2 switches simultaneously

TABLE 3.7-3 (Continued)

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
STEAMLINE ISOLATION (continued)					
b. High containment pressure (Hi-Hi setpoint)	4	3	3		17
c. Manual	1/steamline	1/steamline	1/steamline		21
d. Automatic actuation logic	2	2	1		22
3. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam generator water-level high-high	3/steam generator	2/steam generator	2/in any one steam generator		20
b. Automatic actuation logic and actuation relay	2	2	1		22
c. Safety injection	See Item #1 of Table 3.7-2 (all SI initiating functions and requirements)				

TABLES 3.7-2 AND 3.7-3
TABLE NOTATIONS

ACTION 14. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 30 hours. One channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 15. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 17. With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 6 hours and the Minimum OPERABLE Channels requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1.

ACTION 19. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours.

ACTION 20. With the number of OPERABLE channels one less than the Total Number of Channels, REACTOR CRITICAL and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.1.

TABLES 3.7-2 ANDS 3.7-3 (Continued)
TABLE NOTATIONS

ACTION 21. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirements, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 22. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, be in at least HOT SHUTDOWN within 10 hours and reduce pressure and temperature to less than 450 psig and 350° within the next 8 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 23. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

ACTION 24. With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or reduce pressure and temperature to less than 450 psig and 350°F within the next 12 hours.

ACTION 25. With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the bypassed condition within 6 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1.

TABLE 3.7-4

ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

No.	Functional Unit	Channel Action	Setting Limit
1	High Containment Pressure (High Containment Pressure Signal)	a) Safety Injection b) Containment Vacuum Pump Trip c) High Press. Containment Isolation d) Safety Injection Containment Isolation e) F.W. Line Isolation	≤ 5 psig
2	High-High Containment Pressure (High-High Containment Pressure Signals)	a) Containment Spray b) Recirculation Spray c) Steam Line Isolation d) High-High Press. Containment Isolation	≤ 10.3 psig
3	Pressurizer Low-Low Pressure	a) Safety Injection b) Safety Injection Containment Isolation c) F.W. Line Isolation	≥ 1,700 psig
4	High Differential Pressure Between Steam Line and the Steam Line Header	a) Safety Injection b) Safety Injection Containment Isolation c) F.W. Line Isolation	≤ 150 psig
5	High Steam Flow in 2/3 Steam Lines	a) Safety Injection	≤ 40% (at zero load) of full steam flow ≤ 40% (at 20% load) of full steam flow ≤ 110% (at full load) of full steam flow
		b) Steam Line Isolation c) Safety Injection Containment Isolation d) F.W. Line Isolation	
	Coincident with Low T_{avg} or		≥ 541°F T_{avg}
	Low Steam Line Pressure		≥ 500 psig steam line pressure

TABLE 3.7-4

ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

No.	Functional Unit	Channel Action	Setting Limit
6	AUXILIARY FEEDWATER		
	a. Steam Generator Water Level Low-Low	Aux. Feedwater Initiation S/G Blowdown Isolation	≥ 5% narrow range
	b. RCP Undervoltage	Aux. Feedwater Initiation	≥ 70% nominal
	c. Safety Injection	Aux. Feedwater Initiation	All S.I. setpoints
	d. Station Blackout	Aux. Feedwater Initiation	≥ 46.7% nominal
	e. Main Feedwater Pump Trip	Aux. Feedwater Initiation	N.A.
7	LOSS OF POWER		
	a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	Emergency Bus Separation and Diesel start	75 (±1.0)% volts with a 2 (+5, -0.1) second time delay
	b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	Emergency Bus Separation and Diesel start	90 (±1.0)% volts with a 60 (± 3.0) second time delay (Non CLS, Non SI) 7 (± .35) second time delay (CLS or SI Conditions)
8	NON-ESSENTIAL SERVICE WATER ISOLATION		
	a. Low Intake Canal Level	Isolation of Service Water flow to non-essential loads	23 feet-6 inches
9	RECIRCULATION MODE TRANSFER a. RWST Level-Low	Initiation of Recirculation Mode Transfer System	≥ 18.93% ≤ 19.43%

TABLE 3.7-5

AUTOMATIC FUNCTIONS
OPERATED FROM RADIATION MONITORS ALARM

Monitor Channel	Automatic Function At Alarm Conditions	Monitoring Requirements	Alarm Setpoint µCi/cc
1. Component cooling water radiation monitors	Shuts surge tank vent valve HCV-CC-100	See Specification 3.13	Twice Background
2. Containment particulate and gas monitors (RM-RMS-159 & RM-RMS-160, RM-RMS-259 & RM-RMS-260)	Trips affected unit's purge supply fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)	See Specificatoin 3.10	Particulate $\leq 9 \times 10^{-9}$ Gas $\leq 1 \times 10^{-5}$
3. Manipulator crane area monitors (RM-RMS-162 & RM-RMS-262)	Trips affected unit's purge supply fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)	See Specification 3.10	≤ 50 mrem/hr

TABLE 3.7-5(a)

EXPLOSIVE GAS MONITORING INSTRUMENTATION

Instrument	Total No. of Channels	Minimum OPERABLE Channels	Action
1. Waste Gas Holdup System Explosive Gas Monitoring System Oxygen Monitor	1	1	1

ACTION 1 - With the number of channels OPERABLE less than required by the minimum OPERABLE channels requirement, operation of this waste gas holdup system may continue provided grab samples are collected (1) at least once per 4 hours during degassing operations to the waste gas decay tank and (2) at least once per 24 hours during other operations. Samples shall be analyzed within 4 hours after collection.

TABLE 3.7-6

ACCIDENT MONITORING INSTRUMENTATION

	Instrument	Total No. Of Channels	Minimum OPERABLE Channels
1.	Auxiliary Feedwater Flow Rate	1 per S/G	1 per S/G
2.	Inadequate Core Cooling Monitor		
	a. Reactor Vessel Coolant Level Monitor	2	1
	b. Reactor Coolant System Subcooling Margin Monitor	2	1
	c. Core Exit Thermocouples	2 (Note 2)	1 (Note 2)
3.	PORV Position Indicator	2/valve	1/valve
4.	PORV Block Valve Position Indicator	1/valve	1/valve
5.	Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve
6.	Safety Valve Position Indicator (Backup Detector)	1/valve	0
7.	Containment Pressure	2	1
8.	Containment Water Level (Narrow Range)	2	1
9.	Containment Water Level (Wide Range)	2	1
10.	Containment High Range Radiation Monitor	2	1 (Note 1, b and c only)
11.	Process Vent High Range Effluent Monitor	2	2 (Note 1, a, b, and c)
12.	Ventilation Vent High Range Effluent Monitor	2	2 (Note 1, a, b, and c)
13.	Main Steam High Range Radiation Monitors (Units 1 and 2)	3	3 (Note 1, a, b, and c)
14.	Aux. Feed Pump Steam Turbine Exhaust Radiation Monitor	1	1 (Note 1, a, b, and c)

Note 1: With the number of operable channels less than required by the Minimum OPERABLE Channels requirements
- a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours
- b. Either restore the inoperable channel to operable status within 7 days of the event, or
- c. Prepare and submit a Special Report to the commission pursuant to specification 6.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable.

Note 2: A minimum of 2 core exit thermocouples per quadrant are required for the channel to be operable.

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS, AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S(1,2) Q(3)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position 3) For the control banks, the bench-board indicators shall be checked against the output of the bank overlap unit
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.	Recirculation Mode transfer				
	a. Refueling Water Storage Tank Level-Low	S	R	M	
	b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	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.	Boric Acid Control	N.A.	R	N.A.	
19.	Containment Sump Level	N.A.	R	N.A.	
20.	Accumulator Level and Pressure	S	R	N.A.	
21.	Containment Pressure-Vacuum Pump System	S	R	N.A.	
22.	Steam Line Pressure	S	R	M	

TABLE 4.1-2A (CONTINUED)

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
18.	Primary Coolant System	Functional	1. Periodic leakage testing (a) on each valve listed in Specification 3.1.C.7a shall be accomplished prior to entering power operation condition after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in cold shutdown condition for 72 hours if testing has not been accomplished in the preceeding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
19.	Containment Purge MOV Leakage	Functional	Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval (c)	
20.	Containment Hydrogen Analyzers	a. Channel Check b. Channel Functional Test c. Channel Calibration using sample gas containing: 1. One volume percent (± 0.25%) hydrogen, balance nitrogen 2. Four volume percent (± 0.25%) hydrogen, balance nitrogen 3. Channel calibration test will include startup and operation of the Heat Tracing System	Once per 12 hours Once per 31 days Once per 92 days on staggered basis	
21.	RCS Flow	Flow $\geq$ 273,000 gpm	Once per refueling cycle	14
22.	RWST parameters	a. Temperature $\leq$ 45°F b. Volume $\geq$ 387,100gallons	Once per shift Once per shift	

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

* See Specification 4.1.D

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

Pursuant to 10 CFR 50.90, by letter dated August 7, 1992, the Virginia Electric and Power Company (the licensee) proposed changes to the Technical Specifications (TS) for the Surry Power Station, Units 1 and 2, respectively. The changes would (1) add explicit operability, action, and surveillance requirements for the Recirculation Mode Transfer (RMT) function, (2) delete the references to Refueling Water Storage Tank (RWST) maximum volume requirements and clarify RWST water temperature requirements, and (3) specify time limits to accomplish unit shutdowns under certain circumstances and to take corrective actions to address minor deviations from TS limits on RWST volume, temperature, or boron concentration. In addition, administrative changes have been incorporated for consistency with NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Revision 4. The changes include the capitalization of TS "defined" words and the reformatting of tables.

2.0 PROPOSED TECHNICAL SPECIFICATION CHANGES

TS 3.3.A.1 is being revised to specify a maximum RWST water temperature of 45° F.

TS 3.3.B is being revised to add time limits, consistent with TS 3.0.1, for placing a unit in hot shutdown and then in cold shutdown.

TS 3.3.B.10 is being added to provide a one-hour TS Action Statement to return RWST volume, temperature, or boron concentration to allowable values prior to entering a 6-hour TS Action Statement to shut down the unit.

TS 3.4.A.3 is being revised to delete the maximum RWST volume and to specify a maximum RWST water temperature of 45° F.

TS 3.4.B.5 is being added to provide a one-hour TS Action Statement to return RWST volume, temperature, or boron concentration to allowable values prior to entering a 6-hour TS Action Statement to shut down the unit.

TS 3.4.B is being revised to add time limits, consistent with TS 3.0.1 and NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Revision 4, for placing a unit in cold shutdown within 30 hours if safety injection equipment cannot be returned to service within 48 hours after reaching hot shutdown.

TS 3.4.C is being deleted to be consistent with the previous deletions of the provision which permitted operation with RWST water temperature greater than 45° F.

TS Table 3.7-2 is being revised to add Item No. 7, Recirculation Mode Transfer (RMT). Operability requirements for the RWST level channels and actuation logic channels are being specified. A corresponding TS Action Statement (Action 25) is being included which will require an inoperable RWST level channel to be placed in bypass within 6 hours with the total number of operable channels one less than the total number of channels or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. In addition, the Action Statement allows one additional channel to be bypassed for up to 4 hours for surveillance testing.

TS Table 3.7-4 is being revised to add Item No. 9, Limiting Instrument Settings for RMT actuation.

TS 3.7 Basis Section for Accident Monitoring Instrumentation is being revised to accurately reflect valve position indication devices for Pressurizer Power Operated Relief Valves (PORV). The PORV position indication operability requirements are being restated consistent with NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Revision 4.

TS Table 4.1-1 is being revised to clearly require RWST level instrumentation channel functional testing and actuation logic testing on a monthly basis.

TS Table 4.1-2A is being revised to add surveillance requirements for RWST water temperature and volume.

In addition, administrative changes are being completed to achieve consistency throughout the TS.

3.0 EVALUATION

The proposed changes, as specified above, address two major safety considerations. They are (1) the effect on RMT reliability with a failed RWST level instrument in the bypass condition and (2) the effect on RWST integrity with the deletion of the maximum volume requirement. The remainder of the proposed changes are administrative in nature and do not significantly affect plant operations.

Bypassing a failed RWST level instrument changes the RMT actuation logic from two-out-of-four channels to two-out-of-three channels sensing a low RWST level. With a level instrument in bypass, the capability to withstand a single active failure is still maintained. Continued plant operation with an RWST instrument in bypass is consistent with NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Revision 4. The existing TS do not explicitly provide operability requirements for RMT or the RWST level instruments. Therefore, a failed RWST level instrument would require entering TS 3.0.1, shutting down the unit and placing the plant in an unnecessary transient. Conversely, placing a failed instrument in trip would increase the probability of a spurious RMT actuation which could cause loss of charging pump suction during normal operation or loss of safety injection suction during a design basis accident. Allowing continued plant operation with an RWST level instrument in bypass does not significantly change the level of protection afforded by the RMT function, while avoiding unnecessary plant transients.

A seismic evaluation was done for the RWST and its contents which established a maximum volume limit during a 1981 containment spray system modification. A subsequent analysis was performed to confirm the seismic adequacy of the tank for any maximum volume. Modifications were made to the RWST support system as a result of this analysis. The analysis concluded that the tank design was adequate throughout the range of RWST volume. With the implementation of these modifications, specifying a maximum RWST volume is no longer necessary.

4.0 SUMMARY

The staff has reviewed the licensee's proposed revisions to TS Sections 1, 3.3, 3.4, 3.7, and 4.1 and finds them to be acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

6.0 ENVIRONMENTAL CONSIDERATIONS

Pursuant to 10 CFR 51.32 an environmental assessment has been published (58 FR 28423) in the Federal Register on May 13, 1993. Accordingly, the Commission has determined that the issuance of this amendment will not result in any environmental impact other than those evaluated in the Final Environmental Statement.

7.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: T. Farnholtz

Date: July 8, 1993

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 180 to Facility Operating License No. DPR-32 and Amendment No. 180 to Facility Operating License No. DPR-37, issued to the Virginia Electric and Power Company (the licensee), which revised the Technical Specifications for operation of the Surry Power Station, Units 1 and 2, located in Surry County, Virginia. These amendments are effective as of the date of issuance.

The amendments modified the Technical Specifications to establish operating requirements for the recirculation mode transfer function (RMT) and include the following: 1) Limiting Conditions for Operation, Action Statements, and Surveillance Requirements for the RMT function, 2) deletion of refueling water storage tank maximum volume requirements and clarification of refueling water storage tank water temperature requirements, and 3) administrative changes to provide consistency throughout.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on December 10, 1992 (57 FR 58522). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment (58 FR 28423).

For further details with respect to the action see (1) the application for amendments dated August 7, 1992, (2) Amendment No. 180 to License No. DPR-32 and Amendment No. 180 to License No. DPR-37, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at the Swem Library, College of William and Mary, Williamsburg, Virginia 23185. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Document Control Desk.

Dated at Rockville, Maryland this 9th day of July 1993.

FOR THE NUCLEAR REGULATORY COMMISSION

Bart C. Buckley

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Project Directorate I-4
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