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Docket Nos. 50-280
and 50-281

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

Dear Mr. Ferguson:

The Commission has issued the enclosed Amendment Nos. 56 and 55 to Facility Operating License Nos DPR-32 and DPR-37 for the Surry Power Station, Unit 1 and 2. These amendments are in response to your application dated October 14, 1977, as supplemented.

The amendments consist of additions to the Technical Specifications which incorporate limiting conditions for operation and surveillance requirements for the low temperature overpressure protection system.

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

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 56 to DPR-32
2. Amendment No. 55 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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

March 4, 1980

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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. 56
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 October 14, 1977, as supplemented, 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 the 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 Appendices A and B, as revised through Amendment No. 56 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 4, 1980

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. 55
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 October 14, 1977, as supplemented, 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 the 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 Appendices A and B, as revised through Amendment No. 55 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 4, 1980

ATTACHMENT TO LICENSE AMENDMENT NOS.56 AND 55

FACILITY OPERATING LICENSE NOS. DPR-32 AND DPR-37

DOCKET NOS. 50-280 AND 50-281

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

Remove	Insert
3.1-1	3.1-1
3.1-2	3.1-2
3.1-23	3.1-23 3.1-24 3.1-25
4.1-8	4.1-8
--	4.1-9a
6.6-16a	6.6-16a

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objectives

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

These conditions relate to: operational components, heatup and cooldown, leakage, reactor coolant activity, oxygen and chloride concentrations, minimum temperature for criticality, and reactor coolant system overpressure mitigation.

A. Operational Components

Specifications

1. Reactor Coolant Pumps

   a. A reactor shall not be brought critical with less than two pumps, in non-isolated loops, in operation.

   b. If an unscheduled loss of one or more reactor coolant pumps occurs while operating below 10% rated power (P-7) and results in less than two pumps in service, the affected plant shall be shutdown and the reactor made subcritical by inserting all control banks into the core. The shutdown rods may remain withdrawn.

c. A minimum of one pump in a non-isoalted loop, or one residual heat removal pump and its associated flow path, shall be in operation during reactor coolant boron concentration reduction.

d. Reactor power shall not exceed 50% of rated power with only two pumps in operation unless the overtemperature ΔT trip setpoints have been changed in accordance with Section 2.3, after which power shall not exceed 60% with the inactive loop stop valves open and 65% with the inactive loop stop valves closed.

e. When all three pumps have been idle for > 15 minutes, the first pump shall not be started unless: (1) a bubble exists in the pressurizer or (2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

2. Steam Generator

A minimum of two steam generators in non-isolated loops shall be operable when the average reactor coolant temperature is greater than 350°F.

3. Pressurizer Safety Valves

a. One valve shall be operable whenever the head is on the reactor vessel, except during hydrostatic tests.

References

(1) FSAR 4.2

(2) FSAR 9.2

G. Reactor Coolant System Overpressure Mitigation

Specification

1. The Reactor Coolant system overpressure mitigating system shall be operable as described below.

   a. Whenever the reactor coolant average temperature is greater than 350°F, a bubble shall exist in the pressurizer with the necessary sprays and heaters operable.

   b. Whenever the reactor coolant average temperature is ≤ 350°F and the reactor vessel head is bolted:

      (1) A maximum of one charging pump operable.

      (2) Two charging pumps shall be demonstrated inoperable at least once per 12 hours by verifying the motor circuit breakers have been removed from their power supply or the benchboard control switch is in the "PULL-TO-LOCK" position.

      (3) Two operable Power Operated Relief Valves (PORV) with a lift setting of ≤ 435 psig, or

      (4) A bubble in the pressurizer with a maximum pressurizer narrow range level of 33%. After a period of 72 hours, two PORV's must also be operable, or

      (5) The Reactor Coolant system vented through one opened PORV, or an equivalent size opening.

2. The requirements of Specification 3.1.G.1.b may be modified as follows:

   a. One PORV may be inoperable for a period not to exceed 7 days. If the inoperable PORV is not restored to operable status within 7 days, then depressurize the RCS and open one PORV within the next 8 hours.

b. With both PORV's inoperable, depressurize the RCS within 8 hours unless specification 3.1.G.1.b.(4) is in effect. When the RCS has been depressurized, open one PORV or establish the conditions listed below. Maintain the RCS depressurized until both PORV's have been restored to operable status.

(1) A maximum Pressurizer narrow range level of 33%.

(2) The series RHR inlet valves opened and their respective breakers locked open or an alternate letdown path operable.

(3) Limit charging flow to less than 150 gpm.

(4) Safety Injection accumulator discharge valves closed and their respective breakers locked open.

c. When the conditions noted in 3.1.G.2.b.(1) through 3.1.G.2.b.(4) above are required to be established, their implementation shall be verified at least once per 12 hours.

3. In the event that the Reactor Coolant System Overpressure Mitigating System is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.6.4 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the Mitigating system or the administrative controls on the transient and any corrective actions necessary to prevent recurrence.

<u>Basis</u>

The operability of two PORV's or the RCS vented through an opened PORV ensures that the Reactor Vessel will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when the Reactor Coolant average temperature is $\leq$ 350°F and the Reactor Vessel Head bolted. When the Reactor Coolant average temperature is > 350°F overpressure protection is provided by a bubble in the pressurizer and/or pressurizer safety valves. A single PORV has adequate relieving

capability to protect the Reactor Vessel from overpressurization when the transient is limited to either (1) the start of an idle Reactor Coolant Pump with the secondary water temperature of a steam generator $\leq$ 50°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.
The limitation for a maximum of one charging pump allowed operable and the surveillance required to verify that two charging pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV, or equivalent.

A maximum Pressurizer narrow range level of 33% has been selected to provide sufficient time, approximately 10 minutes, for operator response in case of a malfunction resulting in maximum charging flow from one charging pump (600 gpm). Operator action would be initiated by at least two alarms that would occur between the normal operating level and the maximum allowable level (33%). When both PORV are inoperable and it is impossible to manually open at least one PORV, additional administrative controls shall be implemented to prevent a pressure transient that would exceed the limits of Appendix G to 10 CFR Part 50.

The requirements of this specification are only applicable when the Reactor Vessel head is bolted. When the Reactor Vessel head is unbolted, a RCS pressure of < 100 psig will lift the head, thereby creating a relieving capability equivalent to at least one PORV.

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	
33.	Reactor Coolant Pressure (Low)	N.A.	R	N.A.	

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

TABLE 4.1.2A (CONTINUED)

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
16.	Reactor Vessel Overpressure Mitigating System (except backup air supply)	Functional & Setpoint	Prior to decreasing RCS temperature below 350°F and monthly while the RCS is <350°F and the Reactor Vessel Head is bolted.	None
17.	Reactor Vessel Overpressure Mitigating System Backup Air Supply	Setpoint	Refueling	None

c. With no fire suppression water system operable, within 24 hours; notify the Commission outlining the action taken and the plans and schedule for restoring the system to operable status.

d. With redundant fire suppression water system component inoperable for more than 14 days, submit a Special Report to the Commission within the next 10 days outlining the cause of inoperability and the plans for restoring the component to operable status.

e. With the CO_2 fire protection system inoperable for more than 14 days, submit a Special Report to the Commission within the next 10 days outlining the cause of inoperability and the plans for restoring the system to operable status.

f. With the Records Vault halon fire protection system inoperable for more than 14 days, submit a Special Report to the Commission within the next 10 days outlining the cause of inoperability and the plans for restoring the system to operable status.

g. In the event that the Reactor Vessel Overpressure Mitigating System is used to mitigate a RCS pressure transient, submit a Special Report to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or the administrative controls on the transient and any corrective action necessary to prevent recurrence.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 56 AND 55 TO

FACILITY OPERATING LICENSE NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By letter to the Virginia Electric and Power Company (the licensee) dated August 11, 1976, the NRC requested an evaluation of the Surry Power Station, Unit Nos. 1 and 2, system designs to determine susceptibility to overpressurization events, an analysis of the possible events and proposed interim and permanent modifications of systems and procedures to reduce the likelihood and consequences of such events. By letter dated October 14, 1977 (Reference 1) which supplements other letters (References 4-12), the licensee submitted the information we requested including the administrative operating procedures, the proposed low temperature overpressure protection system (OPS), and proposed changes to the Technical Specifications. The proposed OPS includes sensors, actuating mechanisms, alarms, and valves to prevent a reactor coolant system transient from exceeding the pressure/temperature limits specified in the Surry Units 1 and 2 Technical Specifications as required by Appendix G to Chapter 10, Code of Federal Regulations, Part 50 (10 CFR 50).

Background

Over the last few years, incidents identified as pressure transients have occurred in pressurized water reactors. The term "pressure transients," as used in this report, refers to events during which the temperature pressure limits of the reactor vessel, as shown in the facility Technical Specifications, are exceeded. All of these incidents occurred at relatively low temperature (less than 200°F) where the reactor vessel material toughness (resistance to brittle failure) is reduced.

The "Technical Report on Reactor Vessel Pressure Transients" in NUREG-0138 (Reference 2) summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to resolve this issue by reducing the likelihood of future pressure transient events at operating reactors. A brief discussion is presented here.

Reactor vessels are constructed of high quality steel made to rigid specifications, and fabricated and inspected in accordance with the time-proven rules of the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at reactor operating conditions. However, since reactor vessel steels are less tough and could possibly fail in a brittle manner if subjected to high pressures at low temperatures, power reactors have always operated with restrictions on the pressure allowed during startup and shutdown operations.

At operating temperatures, the pressure allowed by Appendix G limits is in excess of the setpoint of currently installed pressurizer code safety valves. However, most operating PWRs did not have pressure relief devices to prevent pressure transients during cold conditions from exceeding the Appendix G limit.

By letter dated August 11, 1976 (Reference 3), we requested that the licensee begin efforts to design and install plant systems to mitigate the consequences of pressure transients at low temperatures. It was also requested that operating procedures be examined and administrative changes be made to guard against initiating overpressure events. We felt that proper administrative controls were required to assure safe operation for the period of time prior to installation of the proposed overpressure mitigating hardware.

The licensee responded (References 4, 5, and 6) with preliminary information describing interim measures to prevent these transients along with some discussion of proposed hardware. The proposed hardware change was to install a low pressure actuation setpoint on the pressurizer air-operated relief valves.

The licensee participated as a member of a Westinghouse user's group which was formed to support the analysis effort required to verify the adequacy of the proposed system to prevent overpressure transients. Using input data generated by the user's group, Westinghouse performed transient analyses (Reference 10) which were used as the basis for plant-specific analyses.

We requested additional information concerning the proposed procedural changes and the proposed hardware changes. The licensee provided the required responses (References 7 and 8). Reference 1 transmitted the plant-specific analysis for Surry Units 1 and 2.

Through a series of meetings and correspondence with PWR vendors and licensees, we developed a set of criteria for an acceptable overpressure mitigating system.

The proposed overall approach to eliminating overpressure events incorporates administrative, procedural, and hardware controls with reliance upon the plant operator for the principal line of defense. Preventive administrative and procedural measures include (a) explicit procedural precautions, (b) deenergization of essential components not required during the cold shutdown mode of operation, and (c) maintaining a nonwater solid reactor coolant system condition whenever possible.

The basic design criteria that were applied in determining the adequacy of the electrical, instrumentation, and control aspects of the low temperature overpressure protection system are:

Operator Action: No credit can be taken for operator action for 10 minutes after the operator is aware of a transient.

Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

Testability: The system must be testable on a periodic basis consistent with the system's employment.

Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

In addition to complying with these criteria, the licensee agreed to provide a variety of alarms to alert the operator to (a) manually enable the pressure protection system during cooldown, (b) indicate the occurrence of a pressure transient, and (c) indicate closure of either power-operated relief valve (PORV) isolation valve which ensures a complete pathway from the pressurizer to the pressurizer relief tank.

Design Basis Events

The incidents that have occurred to date have been the result of operator errors or equipment failures. Two varieties of pressure transients can be identified: a mass input type from charging pumps, safety injection pumps, safety injection accumulators; and a heat addition type which causes thermal expansion from sources such as steam generators or decay heat.

On Westinghouse designed plants, the most common cause of the overpressure transients to date has been isolation of the letdown path. Letdown during low pressure operations is via a flowpath through the RHR system. Thus, isolation of RHR can initiate a pressure transient if a charging pump is left running. Although other transients occur with lower frequency, those which result in the most rapid pressure increases were identified by the staff for analysis. The most limiting mass input transient identified by the staff is inadvertent injection by the largest safety injection pump. The most limiting thermal expansion transient is the start of a reactor coolant pump with a 50°F temperature difference between the water in the reactor vessel and the water in the steam generator.

Based on the historical record of overpressure transients and the imposition of more effective administrative controls, we believe that the limiting events identified above form an acceptable basis for analyses of the proposed overpressure mitigating system.

Evaluation

System Description

The licensee adopted the "Reference Mitigating System" developed by Westinghouse and the user's group. The licensee proposed to modify the actuation circuitry of the existing air-operated pressurizer relief valves to provide a low pressure setpoint during startup and shutdown conditions. One PORV has a low pressure setpoint at 410 psig and the other at 425 psig. When the reactor vessel is at low temperatures, with the low pressure setpoint selected, a pressure transient is terminated below the Appendix G limit by automatic opening of these relief valves. A manual switch is used to enable and disable the low pressure setpoint of each relief valve. An enabling alarm which monitors system pressure, the position of the enabling switch, and the upstream isolation valve is provided. The system low setpoint is enabled at a pressure of 390 psig during plant cool-down and is disabled at the same pressure during plant heatup. We find the pressurizer relief valves, with a manually enabled low pressure setpoint, to be an acceptable concept for an overpressure mitigating system. Discussion and evaluation of the system proposed by the licensee follows.

Air Supply

The power operated relief valves (PORVs) are spring-loaded-closed, air required to open valves, which are supplied by a control air source. To assure operability of the valves upon loss of control air, a backup air supply is provided. The backup air supply consists of four seismically restrained compressed air bottles (220 psig) for each PORV. Each tank contains enough air for approximately 31 valve openings. A pressure alarm, transmitting to the control room, will be installed to alert the operator when the compressed air pressure has decayed to the point where it will still provide the required number of cycles for 10 minutes. We find the backup air supply to be acceptable.

Operator Action

Operator awareness of the overpressure transient will be by the low temperature overpressure transient alarm. No credit for operator action has been taken until 10 minutes later. We find this acceptable.

Single Failure, Seismic Design, and IEEE Std-279 Criteria

System Electrical and Control Description

The control circuitry for the OPS has been designed to comply with IEEE Std 279-1971, except for the two variations discussed under PORV Channel Separability. The compliance of the design with IEEE 279-1971, including the exceptions described by the licensee (Reference 7), is adequate.

The OPS has two channels that are completely independent except that the channels share an alarm to show that the OPS should be enabled and an alarm to indicate the approach to a possible overpressure event. The alarms are isolated from the channels they serve so that a failure in the alarm circuitry will not incapacitate either channel. Each channel of the OPS is enabled by transferring the key operated ENABLE/DISABLE switch for the channel from the DISABLE to the ENABLE position (two switches must be transferred to completely enable the OPS). Each channel has two pressure setpoints. Setpoint #1 has a value of 400 psig for both channels. When the OPS is enabled, the NDT PRESSURE HIGH annunciator will be activated if the the pressure exceeds Setpoint #1 for either channel, thus alerting the operator of the need for actions to remedy the cause of the increasing pressure. Setpoint #2 has a value of 410 psig for Channel #1 and a value of 425 psig for Channel #2. When the OPS is enabled and the pressure exceeds Setpoint #2 for a channel, the PORV for that channel is opened to provide a pathway from the pressurizer to the pressurizer relief tank.

During power operation the ENABLE/DISABLE switches for both channels of the OPS are in the DISABLE position, and the pressure is above Setpoint #2 for both channels so that the NDT PRESSURE SYSTEM REQUIRED annunciator is off. As the reactor is cooled down the pressure decreases and, when it reaches 400 psig, the NDT PRESSURE SYSTEM REQUIRED annunciator comes on, thus alerting the operator of the need to manually enable the OPS by transferring both key-controlled ENABLE/DISABLE switches to the ENABLE position. If both isolation valves between the pressurizer and the pressurizer relief tank are open, the NDT PRESSURIZER SYSTEM REQUIRED annunciator will go off, indicating that the OPS is enabled. These design features are adequate.

Isolation Valve Alarm. The required isolation valve alarm is provided by the NDT PRESSURE SYSTEM REQUIRED annunciator. When the OPS is being enabled, the annunciator will not clear unless both isolation valves have been opened. This ensures that a path from the pressurizer to the pressurizer relief tank is maintained. With the OPS enabled, the annunciator will alarm upon the closing of either isolation valve. The two channels share a single alarm. These design features are adequate.

PORV Channel Separability. Each of the two PORVs has its own independent instrumentation and control channels, except that the two channels share a common annunciator. A Failure Mode and Effects Analysis (Reference 8) has shown that no single failure can disable both channels, and the licensee has stated that the design meets all of the final criteria except the following two requirements from IEEE 279 for electrical components:

(1) The requirement of automatic removal of a bypass.

The bypass function will be served by two key lock switches, one for each power-operated relief valve, under administrative control. The switch will be enabled at the proper point (temperature versus pressure) on the cooldown curve and disabled at the proper point on the heatup curve. The position of the switch versus system requirements will be annunciated to indicate improper system alignment.

(2) The requirement of identifying components as to protection grade.

The existing components are mounted and wired in control cabinets and wireways. However, channel independence conditions are met, as the channels are totally separate and the new system will also be installed separately. To disrupt the existing system to move the components and wires into protection marked areas does not provide a sufficient advantage to be worth the risk involved to the rest of the station.

The exceptions to IEEE 279 are justified and the design is adequate.

PORV Operation. The pressurizer power-operated relief valves (PORVs) are spring-loaded, normally-shut valves that are opened by motive air controlled by solenoid operated valves (SOVs), one for each PORV when the OPS is enabled. The motive air is normally supplied by the containment instrument air system. To ensure operability upon loss of the normal air supply, each PORV has an independent backup air supply. One PORV opens at 410 psig and resets at 400 psig, the other opens at 425 psig and resets at 415 psig. Each backup air supply will have four high pressure bottles with each bottle capable of opening a PORV 31 times so that the system capacity is 125 cycles. This sizing considers that the fastest system response time is 6 seconds per cycle and that operator response will not take place for 10 minutes. On this basis, the required capacity for the backup air supplies is 100 cycles. Check valves isolate the normal and the two backup air supplies so that a failure in one supply will not disable the other supplies. This design is adequate.

Pressure Transient Reporting and Recording Requirements

The staff position on a pressure transient which causes the overpressure protection system to function, thereby indicating the occurrence of a serious pressure transient, is that it is a 30-day reportable event. In addition, pressure and temperature instrumentation are required to provide a permanent record of the pressure transient. The response times of the temperature/pressure recorders shall be compatible with a pressure transient increasing at a rate of approximately 100 psi per second. This instrumentation shall be operable whenever the OPS is enabled.

Disabling of Essential Components Not Required During Cold Shutdown

Except as required for brief intervals by operating procedures or Technical Specifications, the staff position requires that essential components not required during cold shutdown that could produce an overpressurization event, be disabled or isolated from the RCS during cold shutdown and that the controls to disable or isolate these components be incorporated in the Technical Specifications. In particular, the safety injection accumulators and the high pressure safety injection pumps are included in the components to be disabled or isolated during cold shutdown. While the system is water solid, two of the three charging pumps will be disabled by removal of the power to them. Valves and breakers used to disable equipment during cold shutdown will be tagged or locked to prevent inadvertent changes of state.

System Testability

Testability will be provided prior to establishing a solid system by use of the remotely operated isolation valve, ENABLE/DISABLE switch, and normal electronics surveillance procedure methodology. The testing requirements will be incorporated in the Technical Specifications. The provisions for testability are adequate.

Appendix G

The Appendix G curve submitted by VEPCO for purposes of overpressure transient analysis is the most limiting condition expected over the 40-year life of the plant. The zero degree heatup curve is allowed since most pressure transients occur during isothermal metal conditions. Margins of 60 psig and 10°F are included for possible instrument errors. The Appendix G limit at 100°F according to these conditions is 500 psig. The staff finds that use of this value is acceptable as a basis for overpressure mitigating system performance.

Setpoint Analysis

The one-loop version of the LOFTRAN (Reference WCAP 7907) code was used to perform the mass input analyses. The four loop version was used for the heat input analysis. Both versions require some input modeling and initialization changes. LOFTRAN is currently under review by the staff and is judged to be an acceptable code for treating problems of this type.

The results of this analysis are provided in terms of PORV setpoint overshoot. The predicted maximum transient pressure is simply the sum of the overshoot magnitude and the setpoint magnitude. The PORV setpoints are adjusted so that, given the setpoint overshoot, the resultant pressure is still below that allowed by Appendix G limits.

The licensee relied upon the following Surry Units 1 and 2 plant characteristics to determine the pressure reached for the design basis pressure transients:

SI pump flow rate @ 500 psig	83 lb/sec
RCS volume	10,000 ft^3
SG heat transfer area	58,000 ft^2
Relief valve setpoint	435 psig

The analyses were performed assuming a single PORV setpoint of 435 psig, although the actual setpoints are 410 and 425 psig. Westinghouse also identified certain other assumptions and input parameters as conservative with respect to the analysis. Some of these are listed here.

(1) One PORV was assumed to fail.

(2) The RCS was assumed to be rigid with respect to expansion.

(3) Conservative heat transfer coefficients were assumed for the steam generator.

The staff agrees that most of these are conservative assumptions. It is prudent to assume a PORV failure.

Mass Input Case

The inadvertent start of a safety injection pump with the plant in a cold shutdown condition was selected as the limiting mass input case. For this transient, a relief valve opening time of 1.7 seconds was used. VEPCO has verified that this time is conservative.

Westinghouse provided the licensee with a series of curves based on the LOFTRAN analysis of a generic plant design which indicates PORV setpoint overshoot for this transient as a function of system volume, relief valve opening time, and relief valve setpoint. These sensitivity analyses were then applied to the Surry Units 1 and 2 plant parameters to obtain a conservative estimate of the PORV setpoint overshoot. We find this method of analysis to be acceptable.

Using the Westinghouse methodology, the Surry Units 1 and 2 PORV setpoint overshoot was determined to be 65 psi. With a relief valve setpoint of 435 psig, a final pressure of 500 psig is reached for the worst case mass input transient. Since the Appendix G limit at temperatures above 100°F is above 500 psig, we concluded that the system performance was acceptable with a 435 psig low pressure relief valve setpoint. The actual setpoints of 410 and 425 psig add additional conservatism.

Heat Input Case

Inadvertent startup of a reactor coolant pump with a primary to secondary temperature differential across the steam generator of 50°F, and with the plant in a water solid condition, was selected as the limiting heat input case. For the heat input case, Westinghouse provided the licensee with a series of curves based on the LOFTRAN analysis of a generic plant design to determine the PORV setpoint overshoot as a function of RCS volume, steam generator UA and initial RCS temperature. For this transient, a relief valve opening time of 1.7 seconds was used.

The calculated final pressure for the heat transient for a fixed ΔT of 50°F depends on the initial RCS temperature. The most limiting heat input case resulted in a maximum pressure of 500 psig. Therefore, the Appendix G limits are not exceeded.

We find that the analysis of the limiting mass input and heat input cases shows a maximum pressure transient which does not exceed that allowed by Appendix G limits and is therefore acceptable.

Administrative Controls

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analysis provided by the licensee, a defense-in-depth approach is adopted using procedural and administrative controls. Those specific conditions required to assure that the plant is operated within the bounds of the analysis are spelled out in the Technical Specifications.

Procedures

A number of provisions for the prevention of pressure transients are contained in the Surry Units 1 and 2 operating procedures.

(1) A standing order has been implemented to minimize the period of water solid operation; only fill and vent procedures absolutely require the RCS being maintained in a water solid condition.

(2) To reduce the probability of RCP start from occurring and causing a thermal expansion due to energy transfer from the steam generator, at least one RCP is kept running during cooldowns until the RCS temperature is below 160°F.

(3) A pressurizer steam bubble is maintained prior to any RCP start with the exception of a RCP being started or jogged during the fill and vent procedure.

(4) The RCP operating procedures require a RCS/SG temperature difference of less than 20°F whenever the RCS is water solid.

(5) To assure that the relief capacity of the RHR system (750 gpm at 600 psig) is available to provide RCS pressure relief, the RHR valves are locked open during shutdown operations.

(6) Operation of only one charging pump is permitted during water solid conditions.

(7) The safety injection accumulators are isolated and the safety injection logic is blocked while in a shutdown condition.

We find that the procedural and administrative controls described are acceptable.

Technical Specifications

The licensee has proposed changes to the Technical Specifications to assure operation of the overpressure mitigating system (References 11 and 12). These changes are consistent with the intent of the statements listed below.

(1) Both PORVs must be operable whenever the RCS temperature is less than the minimum pressurization temperature, except one PORV may be inoperable for 7 days. If these conditions are not met, the RCS must be depressurized and vented to the atmosphere or to the pressurizer relief tank within 8 hours.

(2) Operability of the overpressure mitigating system requires that the low pressure setpoint will be selected, the upstream isolation valves open and the backup air supply charged.

(3) No more than one high head SI or charging pump may be energized at RCS temperatures below 350°F, unless the reactor vessel head is removed.

(4) A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer or the SG/RCS temperature difference is less than 50°F.

(5) The overpressure mitigating system must be tested on a periodic basis consistent with the need for its use.

(6) When the plant is in a cold shutdown condition, the safety injection accumulators shall be isolated from the RCS by verifying that the accumulator

isolation valves are in the closed position and power to the valve operators is removed.

Summary

The administrative controls and hardware changes proposed by the licensee provide protection for Surry Units 1 and 2 from the pressure transients at low temperatures by reducing the probability of initiation of a transient and by limiting the pressure of such a transient to below the limits set by Appendix G. We find that the system is acceptable as a long-term solution to the problem of overpressure transients, on the basis that (1) the design complies with the IEEE Std 279-1971 design criteria, (2) the design complies with the seismic design criteria, (3) the system is redundant and meets the single failure criterion, (4) the design requires no operator action for 10 minutes after the operator receives an overpressure action alarm, (5) the system is testable on a periodic basis, and (6) the proposed changes to the Technical Specifications have been reviewed and are in agreement with our requirements.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

References

1. VEPCO letter (Stallings) to NRC (Case) dated October 14, 1977.

2. "Staff Discussion of Fifteen Technical Issues listed in Attachment G November 3, 1976, Memorandum from Director NRR to NRR Staff," NUREG-0138, November 1976.

3. NRC letter (Ziemann) to VEPCO (Stallings) dated August 11, 1976.

4. VEPCO letter (Stallings) to NRC (Rusche) dated September 7, 1976.

5. VEPCO letter (Stallings) to NRC (Rusche) dated November 3, 1976.

6. VEPCO letter (Stallings) to NRC (Rusche) dated December 17, 1976.

7. VEPCO letter (Stallings) to NRC (Rusche) dated February 25, 1977.

8. VEPCO letter (Stallings) to NRC (Rusche) dated April 1, 1977.

9. VEPCO letter (Stallings) to NRC (Case) dated April 22, 1977.

10. "Pressure Mitigating System Transient Analysis Results," prepared by Westinghouse for the Westinghouse user's group on reactor coolant system overpressurization, dated July 1977.

11. VEPCO letter (Stallings) to NRC (Denton) dated October 12, 1978.

12. VEPCO letter (Stallings) to NRC (Denton) dated December 15, 1978.

Dated: March 4, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 56 and 55 to Facility Operating License Nos. DPR-32 and DPR-37 issued to Virginia Electric and Power Company, which revised Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2 (the facilities) located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments consist of additions to the Technical Specifications which incorporate limiting conditions for operation and surveillance requirements for the low temperature overpressure protection system.

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

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

For further details with respect to this action, see (1) the application for amendment dated October 14, 1977 as supplemented (2) Amendment Nos. 56 and 55 to License Nos. DPR-32 and DPR-37, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 4th day of March, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors