May 14, 1993

Docket No. 50-281

DISTRIBUTION
See attached page

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 UNIT 2 - ISSUANCE OF AMENDMENT RE: OPERATION AT REDUCED REACTOR COOLANT SYSTEM NOMINAL OPERATING PRESSURE (TAC NO. M86358)

The Commission has issued the enclosed Amendment No. 177 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated May 6, 1993.

The amendment allows Surry Power Station, Unit 2 to operate at a reactor coolant system nominal operating pressure of 2135 psig for the duration of Cycle 12.

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

Sincerely,

(Original Signed By)
Gus C. Lainas, Assistant Director
for Region II Reactors
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 177 to DPR-37
2. Safety Evaluation

cc w/enclosures:
See next page *Previously Concurred

OFC	:LA:PDII-2	: PM:PDII-2	: D:PDII-2	:SRXB*	:OGC*	:
NAME	:E. Tana	: B. Buckley	: H. Berkow	:R. Jones	:M. Zobler	:
DATE	:05/14/93	:05/14/93	: 05/14/93	:05/13/93	:05/13/93	:

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

Surry Power Station

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DATED: May 14, 1993

AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE NO. DPR-37 - SURRY UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
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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. 177
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 May 6, 1993, 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.

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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. 177, 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, 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: May 14, 1993

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-5	2.1-5
2.2-2	2.2-2
2.3-2	2.3-2
3.3-9	3.3-9
3.12-11	3.12-11

fully withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits dictated by TS Figures 3.12-1A (Unit 1) and 3.12-1B (Unit 2) ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit(3) based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal operating pressure of 2235* psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 60%.

For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also

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

The nominal settings of the power-operated relief valves at 2335* psig, the reactor high pressure trip at 2385** psig and the safety valves at 2485 psig are established to assure never reaching the Reactor Coolant System pressure safety limit. The initial hydrostatic test has been conducted at 3107 psig to assure the integrity of the Reactor Coolant System.

(1) FSAR Section 4

(2) FSAR Section 4.3

* 2235 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operating pressure of 2135 psig.

** ≤ 2310 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operating pressure of 2135 psig.

(b) High pressurizer pressure - $\leq$ 2385* psig.

(c) Low pressurizer pressure - $\geq$ 1860 psig.

(d) Overtemperature T

$$\Delta T \leq \Delta T_o [K_1 - K_2 (\frac{1 + \tau_1 S}{1 + \tau_2 S}) (T - T') + {}_3K (P - P') - f(\Delta I)]$$

where

ΔT_o = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

T' = 574.4°F

P = Pressurizer pressure, psig

P' = 2235 psig

$K_1 = 1.135$	
$K_2 = 0.01072$	
$K_3 = 0.000566$	for 3-loop operation
$K_1 = 0.951$	
$K_2 = 0.01012$	for 2-loop operation with loop stop
$K_3 = 0.000554$	valves open in inoperable loop
$K_1 = 1.026$	
$K_2 = 0.01012$	for 2-loop operation with loop stop
$K_3 = 0.000554$	valves closed in inoperable loop

$\Delta I = q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power

$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Figure 2.3-1

τ_1 = 25 seconds

τ_2 = 3 seconds

(e) Overpower ΔT

$$\Delta T \leq \Delta T_o [K_4 - K_5 (\frac{\tau_3 S}{1 + \tau_3 S}) T - K_6 (T - T') - f(\Delta I)]$$

* $\leq$ 2310 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operating pressure of 2135 psig.

The accumulators (one for each loop) discharge into the cold leg of the reactor coolant piping when Reactor Coolant System pressure decreases below accumulator pressure, thus assuring rapid core cooling for large breaks. The line from each accumulator is provided with a motorized valve to isolate the accumulator during reactor start-up and shutdown to preclude the discharge of the contents of the accumulator when not required. These valves receive a signal to open when safety injection is initiated.

To assure that the accumulator valves satisfy the single failure criterion, they will be blocked open by de-energizing the valve motor operators when the reactor coolant pressure exceeds 1000 psig. The operating pressure of the Reactor Coolant System is 2235* psig and safety injection is initiated when this pressure drops to 600 psig. De-energizing the motor operator when the pressure exceeds 1000 psig allows sufficient time during normal startup operation to perform the actions required to de-energize the valve. This procedure will assure that there is an operable flow path from each accumulator to the Reactor Coolant System during power operation and that safety injection can be accomplished.

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

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.

c. In hot, intermediate and cold shutdown conditions, the step demand counters shall be operable and capable of determining the group demand positions to within ±2 steps. The rod position indicators shall be available to verify rod movement upon demand.

2. If a rod position indicator channel is out of service, then:

a. For operation above 50% of rated power, the position of the RCC shall be checked indirectly using the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod exceeding 24 steps, or

b. Reduce power to less than 50% of rated power within 8 hours. During operations below 50% of rated power, no special monitoring is required.

3. If more than one rod position (RPI) indicator channel per group or two RPI channels per bank are inoperable during control bank motion to achieve criticality or power operations, then the requirements of Specification 3.0.1 will be followed.

F. <u>DNB PARAMETERS</u>

1. The following DNB related parameters shall be maintained within their limits during power operation:

Reactor Coolant System $T_{avg} \leq 578.4°F$
Pressurizer Pressure ≥ 2205* psig
Reactor Coolant System Total Flow Rate ≥ 273,000 gpm

a. The Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once every 12 hours.

* ≥ 2105 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operating pressure of 2135 psig.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177 TO

FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NO. 2

DOCKET NO. 50-281

1.0 INTRODUCTION

By letter dated May 6, 1993, the Virginia Electric and Power Company (the licensee) requested an emergency Technical Specification (TS) change for Surry Power Station, Unit 2. The proposed change would allow operation with a 100 psi reduction in the reactor coolant system (RCS) nominal operating pressure through the end of Cycle 12.

2.0 BACKGROUND

On April 29, 1993, during hydrostatic testing of the Surry, Unit 2 prior to returning to operation following completion of the Cycle 11 refueling outage, RCS leakage past the pressurizer "A" and "C" safety valves was observed. The licensee attributes the leakage to the loss of the water loop seal upstream of the safety valves. The hydrostatic testing was terminated and the RCS pressure reduced to prevent further leakage through the safety valves. To complete the ASME-required hydrostatic test and to prevent further leakage past the safety valves, the licensee found it necessary to mechanically secure (gag) the safety valves. Mechanically securing the leaking pressurizer safety valves was evaluated as an acceptable method to aid in the reseating of the valves and completing the required hydrostatic test without potentially increasing the leakage from the safety valves. The Surry TS requires that all three safety valves be operable with the RCS temperature greater than 350°F. Since the "A" and "C" safety valves would be gagged, and therefore inoperable, the licensee requested and was granted on April 30, 1993, a 36-hour enforcement discretion to perform the hydrostatic test with the "A" and "C" valves gagged with the RCS temperature greater than 350°F. The hydrostatic test was successfully completed; however, loop seals to all the valves could not be established due to continued leakage.

Following completion of the testing described above and with the gags removed and the RCS pressure reduced to normal operating pressure (2235 psig), the "A" safety valve continued to show minor leakage. The RCS pressure was reduced to 1800 psig and slowly repressurized to 2135 psig where the "A" safety valve stabilized. The licensee indicated that despite refurbishing and resetting the lift point, they had experienced difficulty with the same safety valve

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during past start-ups and, therefore, further refurbishing of the "A" valve was not likely to eliminate the leakage. By letter dated May 4, 1993, the licensee requested and was granted an enforcement discretion to operate Surry, Unit 2 at reduced nominal operating pressure (2135 psig) to eliminate leakage until the NRC had received and processed an emergency TS change. By letter dated May 6, 1993, the licensee requested an emergency TS amendment to allow operation of Surry, Unit 2 at an RCS nominal operating pressure of 2135 psig, instead of the normal 2235 psig for the duration of Cycle 12.

3.0 PROPOSED TS CHANGES

Limits on RCS average temperature, pressurizer pressure, and RCS flow are specified to assure that each of the parameters is maintained within the normal, steady-state envelope of operation assumed in the transient and accident analyses. The licensee has stated that these limits have been analytically demonstrated to be adequate to maintain a minimum departure from nucleate boiling ratio (DNBR) which is greater than the design limit throughout each analyzed transient. The proposed changes include a decrease in the specified minimum pressurizer pressure from 2205 to 2105 psig. The effect of this decrease on DNBR margin has been evaluated by the licensee and has been determined to maintain an adequate margin.

Basis Sections of Technical Specifications 2.1, 2.2, and 3.3

The Basis sections of these TS are revised to reflect the change in RCS nominal operating pressure from 2235 to 2135 psig and the reduction of the high pressurizer pressure reactor trip setting limit from 2385 to 2310 psig. In addition, the Basis section of TS 2.3 is revised to reflect a reduction in the power-operated relief valve (PORV) nominal setting from 2335 to 2235 psig. The setpoint change assures that the PORVs will provide a diverse source of RCS overpressure protection prior to reaching the high pressurizer reactor trip setpoint.

Specific changes

Technical Specification 2.3A.2(b) - Instrument Setting Limit for High Pressurizer Pressure Reactor Trip

This specification provides the setting limit for the high pressurizer pressure reactor trip as follows:

(b) High pressurizer pressure - $\leq$ 2385 psig

The proposed changes would modify this item as follows:

(b) High pressurizer pressure - $\leq$ 2385* psig

* $\leq$ 2310 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operating pressure of 2135 psig.

Technical Specification 3.12.F.1 - DNB Parameters

This specification provides for a minimum pressurizer pressure to be maintained during power operation as follows:

Pressurizer Pressure ≥ 2205 psig

The proposed changes would modify this item as follows:

Pressurizer Pressure 2205* psig

* ≥ 2105 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operating pressure of 2135 psig.

Basis Section for Technical Specification 2.1 - Safety Limit Reactor Core

This Basis section states the following:

"...based on steady state nominal operating power levels less than or equal to 100% steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal operating pressure of 2235 psig."

The proposed changes would modify this discussion as follows:

"...based on steady state nominal operating power levels less than or equal to 100% steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal operating pressure of 2235* psig."

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

Basis Section for Technical Specification 2.2 - Safety Limit Reactor Coolant System Pressure

This Basis section states the following:

"The nominal settings of the power-operated relief valves at 2335 psig, the reactor high pressure trip at 2385 psig, and the safety valves at 2485 psig are established to assure never reaching the Reactor Coolant System pressure safety limit."

The proposed changes would modify this discussion as follows:

"The nominal settings of the power-operated relief valves at 2335* psig, the reactor high pressure trip at 2385** psig, and the safety valves at 2485 psig are established to assure never reaching the Reactor Coolant System pressure safety limit."

* 2235 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operating pressure of 2135 psig.

** ≤ 2310 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operating pressure of 2135 psig.

Basis Section for Technical Specification 3.3 - Safety Injection System

This Basis section states the following:

"The operating pressure of the Reactor Coolant System is 2235 psig..."

The proposed changes would modify this discussion as follows:

"The operating pressure of the Reactor Coolant System is 2235* psig..."

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

4.0 DISCUSSION

The major issues considered in the evaluation of the proposed changes were (1) departure from DNBR performance margins, (2) the impact on transient performance parameters other than DNBR, and (3) fuel performance margin.

DNBR Performance Margins

The licensee assessed the impact of the RCS operating pressure reduction on the analyses for DNBR-limited events. For events protected by the overtemperature ΔT (OTΔT) reactor trip, the licensee determined that there is no impact due to the reduced operating pressure. For DNBR-limited events not protected by the OTΔT reactor trip, the licensee determined that the DNBR penalty resulting from the reduced operating pressure would be accommodated by the retained margin in its approved DNBR methodology. The licensee concluded that the results for all DNB-limited events will continue to be bounded by the currently applicable licensing basis analysis for reduced pressure operation.

Non-DNB Accident Performance

In addition to the assessment described above for impact upon events limited by DNBR considerations, the licensee assessed potential indirect impacts from operation at reduced RCS pressure. The assessment of impact from reduced RCS pressure operation resulted in events being placed into one of three classification categories: (1) No Impact, (2) Potentially Impacted, and (3) Impact Requires Quantification. Events in each category are summarized below.

(1) No Impact

From its assessment, the licensee determined that the following events are insensitive to initial RCS pressure conditions, because the acceptance criterion is either not related to RCS pressure or initial pressure does not directly influence transient behavior.

- Uncontrolled Control Rod Assembly Withdrawal from Subcritical Condition

- Chemical and Volume Control System Malfunction (Boron Dilution)
- Startup of an Inactive Reactor Coolant Loop
- Excessive Heat Removal Due to Feedwater System Malfunction
- Excessive Load Increase Incident
- Fuel Handling Accidents
- Volume Control Tank Rupture
- Waste Gas Decay Tank Rupture
- Loss of Coolant Accident (Small and Large Break)
- Rupture of a Main Steam Line

(2) Potentially Impacted

The licensee identified events in this category that either have RCS pressure as the key safety analysis acceptance criterion or have the potential for initial RCS pressure to influence transient behavior during the event. The licensee evaluated these events and concluded that limiting conditions were either insignificantly impacted (or were beneficially impacted) by reduced initial RCS pressure.

- Loss of Normal Feedwater
- Loss of All Alternating Current Power to the Station Auxiliaries
- Steam Generator Tube Rupture
- Rupture of a Control Rod Drive Mechanism Housing (Control Rod Assembly Ejection)
- Main Feedline Break
- Locked Reactor Coolant Pump Rotor (Overpressure Evaluation)

(3) Impact Requires Quantification

The licensee identified one event in this category - the Loss of External Electrical Load. This event establishes the limiting conditions for RCS overpressurization, which is influenced by initial RCS pressure. The licensee stated that previous sensitivities have demonstrated that operation at reduced RCS pressure causes the predicted peak RCS pressure for this event to increase, assuming all other assumptions and protection system actions remain unchanged. The key reactor protection system function which terminates the event is the reactor trip on high pressurizer pressure. Peak RCS pressure is predicted to be greater than in the design analysis for operation at reduced initial pressure, since the primary-secondary energy imbalance exists for a

longer time interval. This causes more energy deposition in the RCS and greater pressurization.

The specific overpressurization event of interest was a complete loss of load transient from an initial pressure of 2105 psig (2235 psi - 100 psi reduction - 30 psi errors) combined with a 75 psi reduction in the assumed high pressurizer pressure reactor trip setpoint. The pressurizer safety valve relief behavior was modeled using the approach described in WCAP-12010, which was recently approved by the NRC. Consistent with the NRC's SER on this methodology, a 1% valve setpoint shift was assumed, since the Surry valves are tested in a steam environment and operate with water loop seals in their installed configuration. An isothermal temperature coefficient of reactivity which bounds the beginning of core life TS limit (+3 pcm/°F) was assumed. The licensee's assessment determined that a peak RCS pressure of 2726 psig would result for this case.

Fuel Performance Assessment

The licensee reviewed the fuel rod design criteria to assess the impact of a reduction in RCS pressure from 2235 psig to 2135 psig for Surry Unit 2 Cycle 12 operation. A reduction in RCS pressure can impact the rod internal pressure evaluation since the differential between the fuel rod internal pressure and the RCS pressure result in an increase in the fuel-to-clad gap. The licensee reanalyzed the rod internal pressure evaluation at the reduced system pressure of 2135 psig and it was determined that all design criteria will continue to be met.

5.0 EVALUATION

Technical Specification 2.3.A.2.(b) - Instrument Setting Limit for High Pressurizer Pressure Reactor Trip

The function of the high pressurizer pressure reactor trip is to protect the RCS and its various components from overpressurization. Reduction of the high pressurizer pressure reactor trip setting limit from 2385 psig to 2310 psig is required to assure that an RCS overpressurization event does not cause RCS pressure to exceed the ASME Section III transient pressure limit of 110% design pressure. Reduction of the setpoint is necessary due to greater pressure overshoot during an overpressure transient. With initial RCS pressure 100 psi below the normal value, peak pressure could exceed the transient limit (2735 psig) unless the high pressurizer pressure trip setpoint is reduced. The licensee determined that the proposed reduction of the setting limit is adequate to provide the required overpressure protection. This result is below the applicable pressure criterion (i.e., ≤110% of design pressure, or 2735 psig), and is therefore acceptable.

Technical Specification 3.12.F.1 - DNB Parameters

Limits on RCS average temperature, pressurizer pressure, and RCS flow are specified to assure that each of the parameters is maintained within the normal, steady state envelope of operation assumed in the transient and accident analysis. These limits have been analytically demonstrated by the

licensee to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. The proposed changes decrease the specified minimum pressurizer pressure from 2205 to 2105 psig. The effect of this proposed decrease on DNBR margin has been evaluated and adequate margin is maintained.

Basis Sections of Technical Specifications 2.1, 2.2 and 3.3

The Bases of these Technical Specifications are revised to reflect the change in RCS nominal operating pressure from 2235 to 2135 psig and the reduction of the high pressurizer pressure reactor trip setting limit from 2385 to 2310 psig. In addition, the Basis Section of Technical Specification 2.3 is revised to reflect a reduction in PORV nominal setting from 2335 to 2235 psig. The setpoint change assures that the PORVs will provide a diverse source of RCS overpressure protection prior to reaching the high pressurizer pressure reactor trip setpoint.

In summary, based on the licensee's reanalysis, the NRC staff concludes that, using the proposed TS values, operation of Surry, Unit 2 will maintain the existing accident analysis margins and there is reasonable assurance that the RCS pressure will be maintained at less than 110 percent of the design accident conditions.

6.0 EMERGENCY CIRCUMSTANCES

10 CFR 50.91(a)(5) requires that whenever an emergency situation exists, a licensee must explain why this emergency situation occurred and why it could not avoid this situation, and the NRC will assess the licensee's reasons for failing to file an application sufficiently in advance of that event. An emergency situation exists when the NRC's failure to act in a timely way would result in derating or shutdown of a nuclear plant, or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level. In such cases, the NRC may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. Our discussion of why this proposed change meets the conditions necessary for emergency consideration is provided below.

Why Emergency Situation Occurred and Could Not be Avoided

On April 29, 1993, hydrostatic testing of the Unit 2 Reactor Coolant System (RCS) was being performed in accordance with Section XI of the ASME Code after modifications to the resistance temperature detector bypass lines and replacement of a safety injection check valve. During the hydrostatic test with an RCS pressure of 2278 psig, indication of discharge from the "A" and "C" pressurizer safety valves was observed. The RCS minor leakage from the "A" and "C" pressurizer safety valves occurred at the hydrostatic test pressure and continued at normal operating pressure.

Technical Specification 3.1.A.3.a requires that all three pressurizer safety valves be operable when the head is installed on the reactor and RCS temperature is above 350°F. Mechanically securing the leaking pressurizer

safety valves was evaluated as an acceptable method to aid in the reseating of the valves and completing the required hydrostatic test without potentially increasing leakage from the safety valves. In order to mechanically secure the safety valves, enforcement discretion from the requirement of TS 3.1.A.3.a was necessary. The enforcement discretion was requested and received on April 30, 1993. This enforcement allowed two safety valves to be mechanically secured for 36 hours.

On May 1, 1993, the "A" and "C" safety valves were mechanically secured and the hydrostatic test was completed. However, when the mechanical securing devices were removed following the test and RCS pressure was returned to normal operating pressure (2235 psig), indications from the pressurizer relief tank, safety valve acoustic monitors, and tail pipe temperature detectors continued to indicate minor leakage of the "A" safety valve. Reducing RCS pressure to approximately 1800 psig and holding at that pressure resulted in the leakage stopping and the associated safety valve temperature profile stabilizing, indicating reformation of the loop seal for the "A" safety valve.

On May 3, 1992, RCS pressure was slowly increased to and held at 2135 psig without any further indication of the leakage phenomenon experienced previously. It was determined then, based on the most recent leakage experience, to request a lowering of RCS operating pressure for Cycle 12 operation as a conservatively prudent action. Operation with a reduced RCS pressure will provide additional margin between valve leakage at power, which can result in valve seat damage, and any consequential plant transient that may result from increased RCS leakage.

Operation of Unit 2 at a reduced RCS pressure of 2135 psig was evaluated and found acceptable. In order to allow reduced pressure operation, changes to the TS are required. Technical Specification 3.12.F.1 requires that RCS pressure be maintained ≥2205 psig during power operation and Technical Specification 2.3.A.2(b) requires that the high pressurizer pressure reactor trip setting be ≤2385 psig. To ensure continued overpressure protection and DNBR margins at the proposed reduced operating pressure, these limits require revision. Discretionary enforcement from the requirements of Technical Specifications 3.12.F.1 and 2.3.A.2.(b) was requested and granted on May 4, 1993 to allow resumption of Unit 2 operation. The licensee submitted a proposed emergency TS change within the provisions of that discretionary enforcement action to allow continued operation of Units 2 at reduced RCS pressure through the end of Cycle 12.

Basis for Emergency TS Change Request

Continued operation of Unit 2 at reduced RCS pressure is proposed as a conservatively prudent action to increase the margin between operating pressure and the onset of valve leakage which has been experienced during the recent hydrostatic testing. Furthermore, the pressurizer safety valves are operable in their present condition and the unit can be safely operated at the proposed reduced RCS pressure. Operation at reduced pressure creates no safety consequences, as discussed in Sections 4.0 and 5.0 above. The safety valves were removed, tested, and refurbished during the recently completed refueling outage. The as-left lift setpoints were within TS limits.

The observed minor safety valve leakage was believed to be the result of differential thermal expansion and the resultant slight misalignment of valve internals. Minor valve discharges can occur until internal stresses are relieved. This minor leakage is not indicative of a setpoint shift or other operability concern. The minor leakage which occurred on Unit 2 has no effect on the safety valve lift pressure or relieving capability. The licensee has concluded that the Unit 2 pressurizer safety valves are operable and that there is no safety or operational reason for placing the unit in cold shutdown for additional valve maintenance. Furthermore, based on the licensee's understanding of the phenomenon, valve maintenance is not likely to preclude a reoccurrence of valve leakage upon subsequent restart. The licensee does not have a spare pressurizer safety valve and the long lead-time to purchase a new valve (approximately 12 to 14 months) would result in a forced shutdown for that time period.

While minor safety valve leakage in itself is not harmful, such leakage can lead to loss of the loop seal and steam cutting of valve seats if allowed to persist. Modifications which will reduce the potential for leakage are scheduled to be implemented during each unit's next refueling outage. The modifications will drain the loop seals, replace the safety valves with valves suitable for steam service, and eliminate the loop seal ovens. The loop seal ovens increase the propensity for leakage by maintaining high safety valve temperatures.

Failure to receive approval of the proposed Technical Specification changes on an emergency basis will increase the risk of RCS leakage and may necessitate a plant shutdown. Based on the recent experience, operation of Unit 2 at the normal RCS pressure of 2235 psig could result in the resumption of minor safety valve leakage.

Continuing leakage would lead to steam cutting of the safety valve seats and increasing RCS leakage. With this concern in mind, the licensee has concluded that operation at a reduced RCS pressure, which provides increased margin between operating pressure and the onset of valve leakage, is both prudent and conservatively safe. Since reduced RCS pressure operation is considered necessary to resume full power operation of Unit 2, this request satisfies the criteria of 10 CFR 50.91(a)(5) for an emergency situation. Without approval of the emergency TS change, a forced shutdown of Surry, Unit 2 for an indefinite period would be required.

7.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The licensee stated that the proposed TS change did not involve a significant hazards consideration, stating that the proposed TS changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. The probability of any accident previously analyzed is not increased. Pressurizer safety valves continue to be operable and capable of performing their intended function. Operating at a reduced RCS pressure does not affect the frequency of accident initiating events. Although the Reactor Coolant System pressure is reduced for one cycle of operation, the departure from nucleate boiling ratio (DNBR) margin is maintained for accidents that challenge the DNBR limits. The loss of load analysis with the modified high pressurizer pressure reactor trip setpoint continues to meet the overpressure design limit. Therefore, there is no increase in the consequences of any accident previously evaluated which is created by operation of Surry Unit 2 at reduced pressure. Furthermore, the proposed changes are being made to provide increased margin between operational pressure and the onset of safety valve leakage. This reduces the potential for valve seat damage and any consequential plant transients that may result from increased RCS leakage.

2. Create the possibility of a new or different kind of accident from any previously evaluated. There are no new failure modes or mechanisms associated with operating Surry Unit 2 at reduced pressure for up to one cycle. Furthermore, the proposed changes are being made to provide increased margin between operational pressure and the onset of safety valve leakage. This reduces the potential for valve seat damage and any consequential plant transient that may result from increased RCS leakage. Therefore, there are no new or different kind of accidents created by operation of Surry Unit 2 at reduced pressure.

3. Involve a significant reduction in the margin of safety. The limiting DNB analyses continue to meet the DNBR acceptance criteria at reduced pressure operation. The applicable overpressure safety analyses acceptance criteria continue to be met with the high pressurizer pressure reactor trip setpoint reduced to less than or equal to 2310 psig. Therefore, the existing margin of safety is safety is not reduced by operation of Surry Unit 2 at reduced pressure.

Based on its evaluation of the above, the staff concurs with the licensee's analysis and, therefore, concludes that this amendment meets the criteria, and does not involve a significant hazards consideration.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official

was notified of the proposed issuance of the amendment. The State official had no comment.

9.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards finding with respect to this amendment. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

10.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) increase the possibility of a new or different kind of accident from any previously evaluated or (c) significantly reduce a safety margin and, therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (3) such activities will be conducted in compliance with the Commission's regulations, and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: May 14, 1993