ATTACHMENT 1

SURRY POWER STATION

PROPOSED TECHNICAL SPECIFICATION CHANGES

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70, and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. <u>Maximum Power Level</u>

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2546 megawatts (thermal).

   B. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 191, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. <u>Reports</u>

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. <u>Records</u>

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 66 and again by Amendment 71

   G. <u>Steam Generator Repair Program</u>

      (1) The Surry Power Steam Generator Repair Program for Unit No. 1 is approved.

      (2) During the steam generator repair program the following conditions shall be met:

         (a) All fuel shall be removed from the reactor pressure vessel and stored in the spent fuel pool.

         (b) Temporary containment and ventilation systems shall be installed and operated for all cutting and grinding operations involving components with removable radioactive contamination greater than 2200 DPM per 100 cm^2 except

L. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved Nuclear Security Personnel Training and Qualifications Program, including amendments and changes made pursuant to 10 CFR 50.54(p). The approved Nuclear Security Personnel Training and Qualifications Program consists of a document withheld from public disclosure pursuant to 10 CFR 2.790(d) identified as "Surry Power Station Nuclear Security Personnel Training and Qualifications Program" dated September 15, 1980. The Nuclear Security Personnel Training and Qualifications Program shall be fully implemented in accordance with 10 CFR 73.55(b)(4), within 60 days of this approval by the Commission. All security personnel shall be qualified within two years of this approval.

M. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittals dated November 5, 1985 (Serial No. 85-136), December 3, 1985 (Serial No. 85-136A), and January 14, 1986 (Serial No. 85-136C).

N. Deleted by Amendment

4. This license is effective as of the date of issuance, and shall expire at midnight on May 25, 2012.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosure Appendix A -
Technical Specifications

Date of Issuance: May 25, 1972

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70, and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2546 megawatts (thermal).

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 191, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. Reports

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 54

   F. Deleted by Amendment 59 and again by Amendment 65

   G. Steam Generator Repair Program

      (1) The Surry Power Steam Generator Repair Program for Unit No. 2 is approved.

      (2) During the steam generator repair program the following conditions shall be met:

         (a) All fuel shall be removed from the reactor pressure vessel and stored in the spent fuel pool.

         (b) Temporary containment and ventilation systems shall be installed and operated for all cutting and grinding operations involving components with removable radioactive contamination greater than 2200 DPM per 100 cm^2 except

L. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved Nuclear Security Personnel Training and Qualifications Program, including amendments and changes made pursuant to 10 CFR 50.54(p). The approved Nuclear Security Personnel Training and Qualifications Program consists of a document withheld from public disclosure pursuant to 10 CFR 2.790(d) identified as "Surry Power Station Nuclear Security Personnel Training and Qualifications Program" dated September 15, 1980. The Nuclear Security Personnel Training and Qualifications Program shall be fully implemented in accordance with 10 CFR 73.55(b)(4), within 60 days of this approval by the Commission. All security personnel shall be qualified within two years of this approval.

M. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittals dated November 5, 1985 (Serial No. 85-136), December 3, 1985 (Serial No. 85-136A), and January 14, 1986 (Serial No. 85-136C).

N. Deleted by Amendment

4. This license is effective as of the date of issuance, and shall expire at midnight on January 29, 2013.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by Roger Boyd/for

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosure Appendix A -
Technical Specifications

Date of Issuance: January 29, 1973

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 2546 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% $\Delta k/k$ and T_{avg} is $\leq$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% $\Delta k/k$ and T_{avg} is $\leq$200°F.

3. INTERMEDIATE SHUTDOWN

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

4. HOT SHUTDOWN

When the reactor is subcritical by at least 1.77% $\Delta k/k$ and T_{avg} is $\geq$ 547°F.

uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the DNB heat flux at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB basis is as follows: there must be at least a 95% probability with 95% confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is based on the entire applicable experimental data set to meet this statistical criterion.[1]

The curves of TS Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent limits equal to, or more conservative than, the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the calculated DNBR is not less than the design DNBR limit or the average enthalpy at the exit of the vessel is equal to the saturation value. The area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would be required if they were based upon the design DNBR limit alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The effects of rod bowing are also considered in the DNBR analyses.

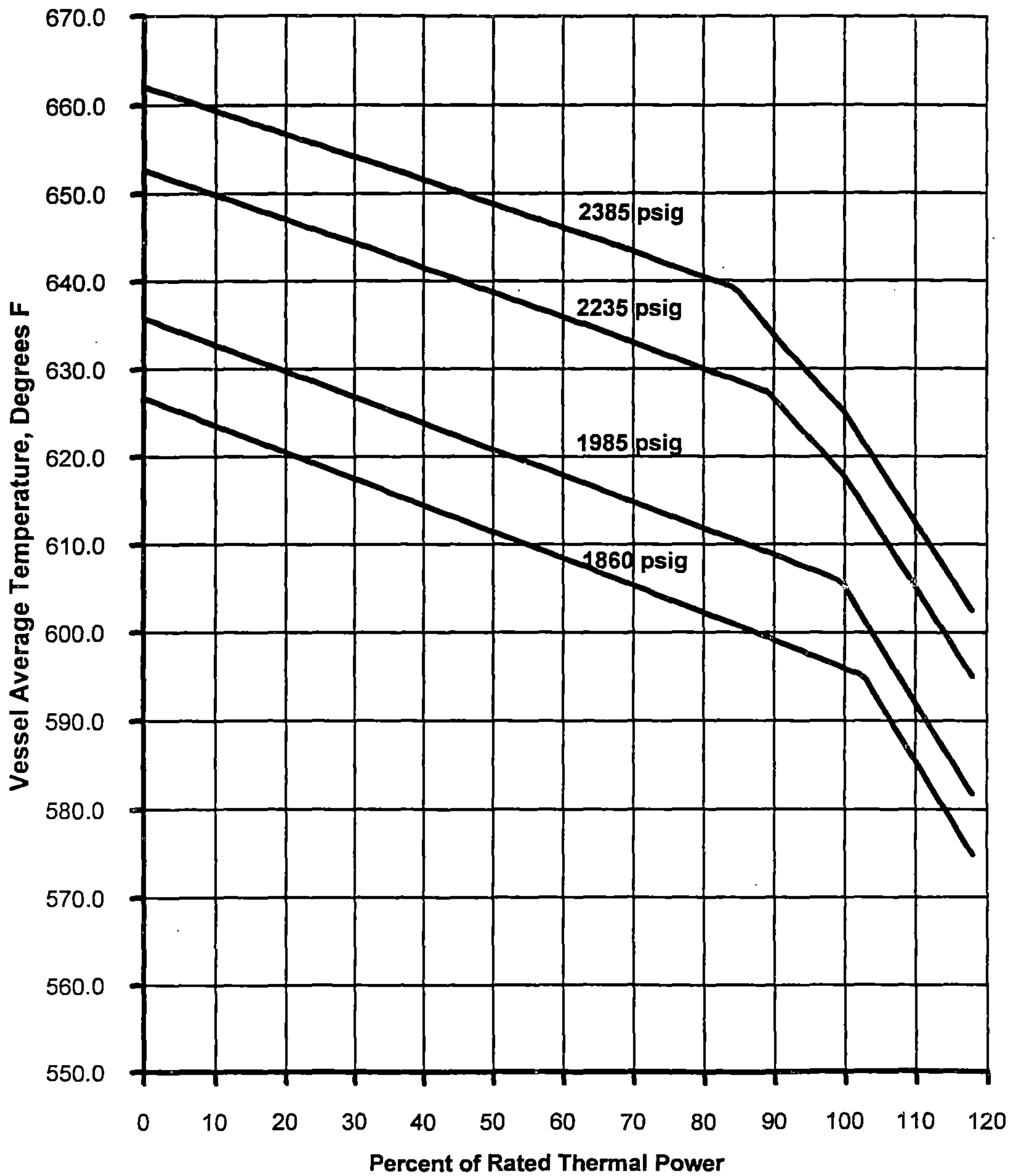
TS Figure 2.1-1 is based on a 1.55 cosine axial flux shape and a statistical treatment of key DNBR analysis parameter uncertainties including an enthalpy rise hot channel factor which follows the following functional form: $F\Delta H(N) = 1.56 [1 + 0.3(1-P)]$ where P is the fraction of RATED POWER. The limits include margin to accommodate rod bowing.[1] TS Figures 2.1-2 and 2.1-3 are based on an $F\Delta H(N)$ of 1.55, a deterministic treatment of key DNB analysis parameter uncertainties, and include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor. The $F\Delta H(N)$ limit presented in the unit- and reload-specific CORE OPERATING LIMITS REPORT is confirmed for each reload to be accommodated by the Reactor Core Safety Limits.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies 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 as specified in the CORE OPERATING LIMITS REPORT 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 573.0°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.

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

TS FIGURE 2.1-1
REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS
THREE LOOP OPERATION, 100% FLOW



Amendment Nos.

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) UFSAR Section 4
2) UFSAR Section 4.3

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

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

(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_0 [K_1 - K_2 (\frac{1 + t_1 S}{1 + t_2 S}) (T - T') + K_3 (P - P') - f(\Delta I)]$$

where

ΔT_o = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

T' = 573.0°F

P = Pressurizer pressure, psig

P' = 2235 psig

K_1 = 1.135

K_2 = 0.01072

K_3 = 0.000566

$\Delta I = q_t - q_b$, 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

t_1 = 25 seconds

t_2 = 3 seconds

(e) Overpower ΔT

$$\Delta T \leq \Delta T_0 [K_4 - K_5 (\frac{t_3 S}{1 + t_3 S}) T - K_6 (T - T') - f(\Delta I)]$$

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The undervoltage reactor trip protects against a decrease in Reactor Coolant System flow caused by a loss of voltage to the reactor coolant pump busses. The underfrequency reactor trip (opens RCP supply breakers and) protects against a decrease in Reactor Coolant System flow caused by a frequency decay on the reactor coolant pump busses. The undervoltage and underfrequency reactor trips are expected to occur prior to the low flow trip setpoint being reached for low flow events caused by undervoltage or underfrequency, respectively. The accident analysis conservatively ignores the undervoltage and underfrequency trips and assumes reactor protection is provided by the low flow trip. The undervoltage and underfrequency reactor trips are retained as back-up protection.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1154 ft^3 of water corresponds to 92% of span. The specified setpoint allows margin for instrument error(7) and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the Auxiliary Feedwater System.(7)

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal unit operations. The prescribed setpoint above which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two or more reactor coolant pumps are lost. Above 50%, an automatic reactor trip will occur if any pump is lost or de-energized. This latter trip

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 three pumps, in non-isolated loops, in operation.

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 loop shall be OPERABLE when the average Reactor Coolant System temperature is greater than 350°F.

3. Pressurizer Safety Valves

a. Three valves shall be OPERABLE when the head is on the reactor vessel and the Reactor Coolant System average temperature is greater than 350°F, the reactor is critical, or the Reactor Coolant System is not connected to the Residual Heat Removal System.

b. Valve lift settings shall be maintained at 2485 psig ± 1 percent

Basis

The specified limit provides protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the following analysis of a steam generator tube rupture accident in UFSAR Chapter 14.3.1.

Rupture of a steam generator tube would allow radionuclides in the reactor coolant to enter the secondary system. The limiting case involves a double-ended tube rupture coincident with loss of the condenser and release of steam from the secondary side to the atmosphere via the main steam safety valves or atmospheric relief valves. This is assumed to continue for 30 minutes in the analysis. The operator will take action to reduce the primary side temperature to a value below that corresponding to the relief or safety valve setpoint. Once this is accomplished the valves can be closed and the release terminated.

Permitting startup and/or REACTOR OPERATION to continue for limited time periods with the reactor coolant's specific activity > 1.0 μCi/cc but < 10.0 μCi/cc DOSE EQUIVALENT I-131 accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER. Although the analysis of a steam generator tube rupture initiated with primary coolant activity at the 10.0 μCi/cc transient limit shows offsite doses well within the 10 CFR 100 limits, operation at the transient limit is restricted to no more than 10 percent of the unit's yearly operating time to limit the risk of appreciable releases following a postulated steam generator tube rupture.

The basis for the 500°F temperature contained in the Specification is that the saturation pressure corresponding to 500°F, i.e., 680.8 psia, is well below the pressure at which the atmospheric relief valves on the secondary side could be actuated.

The accident analysis examines two cases of iodine spiking. For the case with a pre-existing iodine spike, the transient coolant activity limit of 10.0 μCi/cc is assumed. For the case of a concurrent spike, the initial activity is assumed to correspond to the steady state limit of 1.0 μCi/cc. The concurrent iodine spike is modeled with a conservative iodine appearance rate. Both cases show doses at the exclusion area and low population zone boundaries which are well within the 10 CFR Part 100 limits and control room doses which are within the General Design Criterion (GDC) 19 guidelines.

Measurement of $\bar{E}$ will be performed at least twice annually. Calculations required to determine $\bar{E}$ will consist of the following:

1. $\bar{E}$ shall be the average (weighed in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.
2. A determination of the beta and gamma decay energy per disintegration of each nuclide determined in (1) above by applying known decay energies and schemes.
3. A calculation of $\bar{E}$ by appropriate weighing of each nuclide's beta and gamma energy with its concentration as determined in (1) above.

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/cc) 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 either: a) those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", or b) Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Compliance with 10 CFR Part 50, Appendix I."

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 System 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.

2. A minimum of 96,000 gallons of water shall be available in the protected condensate storage tank to supply emergency water to the auxiliary feedwater pump suctions. A minimum of 60,000 gallons of water shall be available in the protected condensate storage tank of the opposite unit to supply emergency water to the auxiliary feedwater pump suction of that unit.

3. All main steam line code safety valves, associated with steam generators in unisolated reactor coolant loops, shall be OPERABLE with lift settings as specified in Table 3.6-1A and 3.6-1B.

C. Prior to reactor power exceeding 10%, the steam driven auxiliary feedwater pump shall be OPERABLE.

D. System piping, valves, and control board indication required for operation of the components enumerated in Specifications 3.6.B.1, 3.6.B.2, 3.6.B.3, and 3.6.C shall be OPERABLE (automatic initiation instrumentation associated with the opposite unit's auxiliary feedwater pumps need not be OPERABLE).

E. The specific activity of the secondary coolant system shall be $\leq$ 0.10 µCi/cc DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant system exceeds 0.10 µCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours.

The capability to supply feedwater to the generators is normally provided by the operation of the Condensate and Feedwater Systems. In the event of complete loss of electrical power to the station, residual heat removal would continue to be assured by the availability of either the steam driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps and the 110,000-gallon protected condensate storage tank. In the event of a fire or high energy line break which would render the auxiliary feedwater pumps inoperable on the affected unit, residual heat removal would continue to be assured by the availability of either the steam driven auxiliary feedwater pump or one of the motor-driven auxiliary feedwater pumps from the opposite unit. A minimum of two auxiliary feedwater pumps are required to be operable* on the opposite unit to ensure compliance with the design basis accident analysis assumptions, in that auxiliary feedwater can be delivered via the cross-connect, even if a single active failure results in the loss of one of the two pumps.

The specified minimum water volume in the 110,000-gallon protected condensate storage tank is sufficient for 8 hours of residual heat removal following a reactor trip and loss of all offsite electrical power. It is also sufficient to maintain one unit at hot shutdown for 2 hours, followed by a 4 hour cooldown from 547°F to 350°F (i.e., RHR operating conditions). If the protected condensate storage tank level is reduced to 60,000 gallons, the immediately available replenishment water in the 300,000-gallon condensate tank can be gravity-fed to the protected tank if required for residual heat removal. An alternate supply of feedwater to the auxiliary feedwater pump suctions is also available from the Fire Protection System Main in the auxiliary feedwater pump cubicle.

The five main steam code safety valves associated with each steam generator have a total combined capacity of 3,842,454 pounds per hour at their individual relieving pressure; the total combined capacity of all fifteen main steam code safety valves is 11,527,362 pounds per hour. The ultimate power rating steam flow is 11,260,000 pounds per hour. The combined capacity of the safety valves required by Specification 3.6 always exceeds the total steam flow corresponding to the maximum steady state power than can be obtained during three reactor coolant loop operation.

* excluding automatic initiation instrumentation

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

The limit on steam generator secondary side iodine - 131 activity is based on limiting the inhalation dose at the site boundary following a postulated steam line break accident to a small fraction of the 10 CFR 100 limits. The accident analysis, which is performed based on the guidance of NUREG-0800 Section 15.1-5, assumes the release of the entire contents of the faulted steam generator to the atmosphere.

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	≥ 11.25% ≤ 15.75%

(3) assuring that environmental conditions will not preclude access to close the valves and 4) that this administrative or manual action will prevent the release of radioactivity outside the containment.

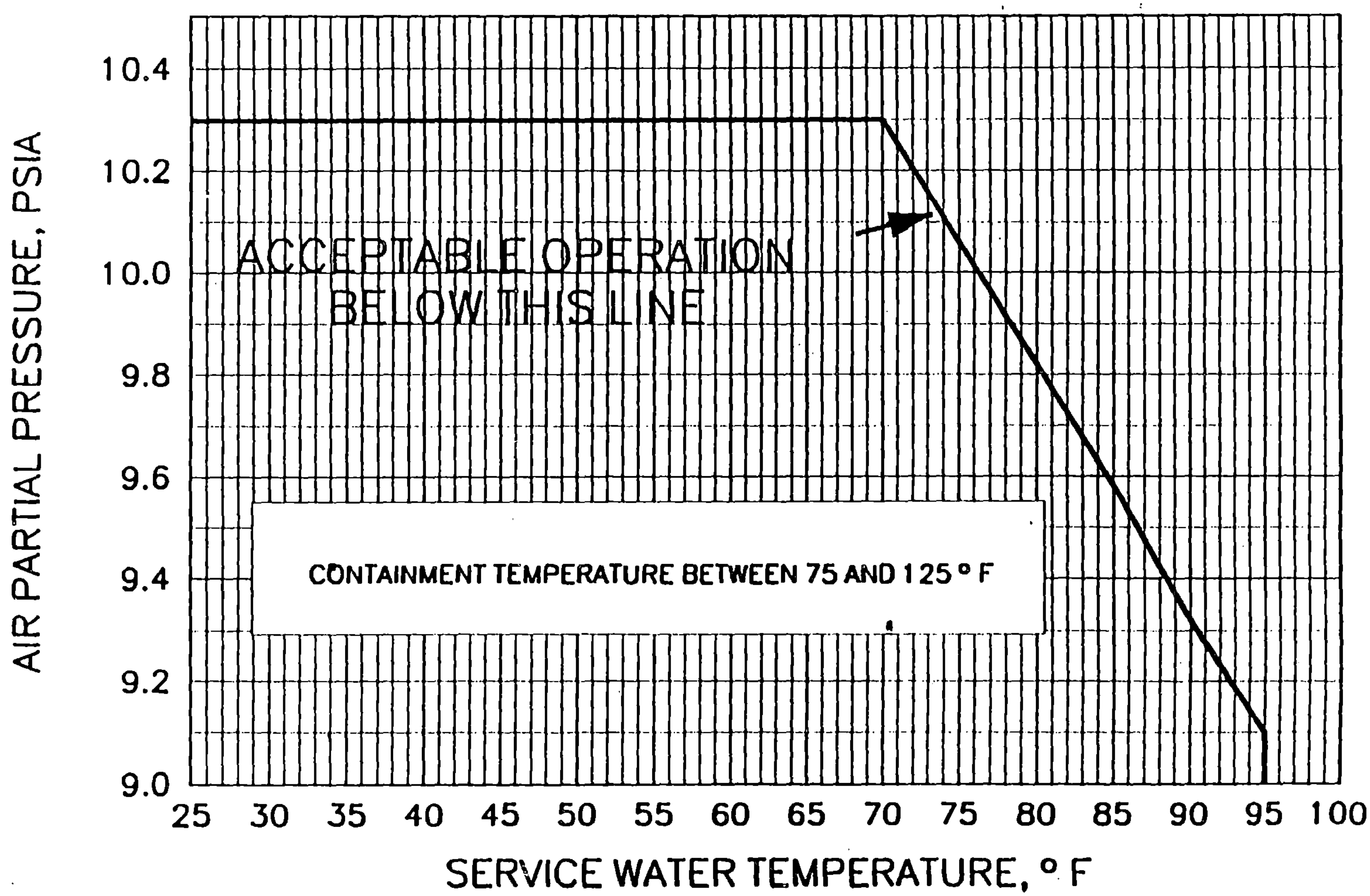
The Reactor Coolant System temperature and pressure being below 350°F and 450 psig, respectively, ensures that no significant amount of flashing steam will be formed and hence that there would be no significant pressure buildup in the containment if there is a loss-of-coolant accident. Therefore, the containment internal pressure is not required to be subatmospheric prior to exceeding 350°F and 450 psig.

The allowable value for the containment air partial pressure is presented in TS Figure 3.8-1 for service water temperatures from 25 to 95°F. The RWST water shall have a maximum temperature of 45°F.

The horizontal limit line in TS Figure 3.8-1 is based on LOCA peak calculated pressure criteria, and the sloped line is based on LOCA subatmospheric peak pressure criteria.

TS FIGURE 3.8-1

SURRY TECHNICAL SPECIFICATION CURVE
MAX CONTAINMENT ALLOWABLE AIR PARTIAL PRESSURE INDICATION VS. SW TEMP



Amendment Nos.

Upon each completion of core loading and installation of the reactor vessel head, specific mechanical and electrical tests will be performed prior to initial criticality.

The fuel handling accident has been analyzed based on the methodology outlined in Regulatory Guide 1.25. The analysis assumes 100% of the gap activity from the highest powered assembly is released after a 100-hour decay period following operation at 2605 MWt.

Detailed procedures and checks insure that fuel assemblies are loaded in the proper locations in the core. As an additional check, the movable incore detector system will be used to verify proper power distribution. This system is capable of revealing any assembly enrichment error or loading error which could cause power shapes to be peaked in excess of design value.

References

UFSAR Section 5.2	Containment Isolation
UFSAR Section 6.3	Consequence Limiting Safeguards
UFSAR Section 9.12	Fuel Handling System
UFSAR Section 11.3	Radiation Protection
UFSAR Section 13.3	Table 13.3-1
UFSAR Section 14.4.1	Fuel Handling Accidents
FSAR Supplement:	Volume I: Question 3.2

3. If more than one rod position indicator channel per group or two rod position indicator channels per bank are inoperable during control bank motion to achieve criticality or POWER OPERATION, then the unit shall be placed in HOT SHUTDOWN within 6 hours.

F. DNB Parameters

1. The following DNB related parameters shall be maintained within their limits during POWER OPERATION:

- Reactor Coolant System T_{avg} ≤ 577.0°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.

b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by measurement at least once per refueling cycle.

2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED POWER within the next 4 hours.

3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED POWER per minute or a THERMAL POWER step increase in excess of 10% of RATED POWER.

(3) $\bar{E}$ determination will be started when the gross gamma degassed activity of radionuclides with half-lives greater than 15 minutes analysis indicates $\geq$ 10μCi/cc. Routine sample(s) for $\bar{E}$ analyses shall only be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was last subcritical for 48 hours or longer.

(4) If the fifteen minute degassed beta and gamma activity is 10% or more of the limit given in Specification 3.6.E, a DOSE EQUIVALENT I-131 analysis will be performed.

(5) When reactor is critical and average primary coolant temperature ≥ 350°F.

(6) Whenever the specific activity exceeds 1.0 μCi/cc DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/cc and until the specific activity of the Reactor Coolant System is restored within its limits.

(7) One sample between 2 & 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED POWER within a one hour period provided the average primary coolant temperature ≥ 350°F.

(8) When the fifteen minute degassed beta and gamma activity is less than 10% of the limit given in Specification 3.6.E.

The containment is designed for a maximum pressure of 45 psig. The containment is maintained at a subatmospheric air partial pressure consistent with TS Figure 3.8-1 depending upon the cooldown capability of the Engineered Safeguards and will not rise above 45 psig for any postulated loss-of-coolant accident.

The initial test pressure for the Type A test is 47.0 psig to allow for containment expansion and equalization. A review was performed to determine the effects of pressurizing containment above its design pressure of 45.0 psig. This review was based on the original containment test at 52 psig. During that test, the calculated stresses were found to be well within the allowable yield strength of the structural reinforcing bars, therefore performance of the Type A test at 47 psig will have no detrimental effect on the containment structure.

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

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

The limitations on closure and leak rate for the containment airlocks are required to meet the restrictions on containment integrity and containment leak rate. Surveillance testing of the airlock seals provides assurance that the overall airlock leakage will not become excessive due to seal damage during the intervals between airlock leakage tests.

<u>References</u>

UFSAR Section 5.4	Design Evaluation of Containment Tests and Inspections of Containment
UFSAR Section 7.5.1	Design Bases of Engineered Safeguards Instrumentation
UFSAR Section 14.5	Loss of Coolant Accident
10 CFR 50 Appendix J	"Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors"

b. A high containment pressure isolation signal closes that automatic trip valves in all normally open lines penetrating the containment which are not required to be open to control containment pressure to perform an orderly reactor shut down without actuation of the consequence limiting safeguards in case of a small Reactor Coolant System leak.

c. A further rise in containment pressure, indicating a major loss-of-coolant accident, produces a containment high-high pressure isolation signal which closes all normally open lines which penetrate the containment which have not been closed by 2-b above.

d. Isolation can be accomplished manually from the control in the Main Control Room is any of the automatic signals fail to actuate the above valves.

C. Containment Systems

1. Following a loss-of-coolant accident, the Containment Spray Subsystems distribute at least 2,600 gpm borated water spray containing sodium hydroxide for iodine removal within the containment atmosphere. The Recirculation Spray Subsystems recirculate at least 3,000 gpm of water from the containment sump.

ATTACHMENT 2

SURRY POWER STATION

DISCUSSION OF CHANGES AND SIGNIFICANT HAZARDS CONSIDERATION

DISCUSSION OF CHANGES

Introduction and Background

Surry Units 1 and 2 are currently licensed for operation at a reactor core power level of 2441 MWt. Virginia Electric and Power Company has undertaken a program to uprate Surry to a maximum reactor core power level of 2546 MWt, approximately a 4.3% increase. The engineering studies supporting the core uprate have been performed in accordance with Westinghouse WCAP-10263, entitled "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor" and dated January 1983.

The Westinghouse methodology for uprating involves the application of the following four key principles to accident analysis evaluation:

- The impact of uprating is verified for compliance with respect to the existing plant licensing basis.
- The impact of uprating is evaluated to determine whether any unreviewed safety questions exist.
- Events are evaluated for adverse impact from the proposed change in power and associated operating conditions. Events which are adversely impacted are reanalyzed.
- Accident reanalyses employ current NRC-approved analytical methods.

As it relates to the system and component design and the NSSS/BOP safety related interfaces, the Westinghouse methodology for uprating involves verification that the systems, components, and interfaces remain in compliance with the design codes, standards, and criteria applied to the current license.

Specific Changes

The Operating License, Technical Specification, and Technical Specification Basis changes required for operation at the uprated conditions can be categorized in the following ways:

- The core rated power change and other changes associated with the revised plant nominal operating conditions.
- Changes reflecting the safety limits and limiting safety system settings associated with revised accident analyses (including containment and radiological consequences analyses).
- Changes in limiting conditions for operation associated with revised analyses or evaluations.

These changes are listed in Section 5.1 of the Surry Core Uprate Licensing Report, contained in Attachment 3.

In addition to those changes, the following administrative changes have been incorporated on the affected pages in this change package:

- Deletion of references to two-loop operation in Technical Specifications 2.3.A.2.(d) (p. TS 2.3-2) and 3.1.A.1.e (p. TS 3.1-3), as well as in various Basis discussions (pages TS 2.1-3, TS 2.1-4, TS 2.1-5, and TS 2.3-7). TS 3.3.A.11 prohibits two-loop power operation.

- Capitalization of defined words and systems (e.g., Reactor Coolant System, OPERABLE, etc.) (Insert A on p. TS 2.3-7, pages TS 2.1-4, TS 3.1-1, TS 3.1-3, TS 3.1-17, TS 3.3-7, TS 3.6-2, and TS 4.1-10A).

- Terminology revisions for consistency (e.g., hot shutdown conditions versus HOT SHUTDOWN, FSAR versus UFSAR, etc.) (pages TS 2.2-2, TS 3.6-2, TS 3.6-4, TS 4.1-10a, and TS 4.4-3).

- Correction of typographical errors (pages TS 2.3-2, TS 3.6-4, TS 3.10-7, and TS 4.1-10a).

Safety Significance

The Surry Core Uprate Project evaluated the ability of plant systems, structures, and components to operate within safe limits, both during normal and accident conditions.

The following plant design bases areas were either evaluated or reanalyzed to assess the effects of uprated operation. Detailed descriptions of the evaluations or reanalyses performed for operation at the proposed uprated conditions are presented in the Surry Core Uprate Licensing Report, contained in Attachment 3.

- Nuclear Design and Core Thermal-Hydraulic Design
- UFSAR Accident Analyses
- Containment Integrity and Safeguards Equipment
- NSSS Accident Radiological Consequences
- Miscellaneous Plant Design Basis Evaluations

The NSSS accident analyses, described in the Surry UFSAR, were evaluated and placed in the following categories:

Events Unaffected by Uprating

- Malpositioning of Part-Length Control Rod Assemblies
- Startup of an Inactive Reactor Coolant Loop
- Likelihood of Turbine-Generator Unit Overspeed

Events Requiring Validation

- Rupture of Main Steam Pipe
- Excessive Heat Removal Due to Feedwater Malfunctions
- Loss of Normal Feedwater
- Rupture of a Control Rod Drive Mechanism Housing
- Small Break Loss of Coolant Accident
- Large Break Loss of Coolant Accident

Events Requiring Reanalysis

- Uncontrolled Control-Rod Assembly Withdrawal from a Subcritical Condition
- Uncontrolled Control-Rod Assembly Withdrawal from Power
- Control-Rod Assembly Drop/Misalignment
- Chemical and Volume Control System Malfunction
- Excessive Load Increase Incident
- Loss of Reactor Coolant Flow
- Locked Rotor Incident
- Loss of External Electrical Load
- Steam Generator Tube Rupture

The NSSS analyses continue to meet applicable acceptance criteria for operation at the uprated conditions.

The revised containment integrity analyses require changes in the limiting condition for operation (LCO) for containment operating conditions. The revised LCO maintains or expands the existing key parameter operating limits.

The following six accidents were analyzed to assess the radiological consequences:

- Loss of Coolant Accident
- Main Steam Line Break
- Steam Generator Tube Rupture
- Locked Rotor Accident
- Fuel Handling Accident
- Waste Gas Decay Tank Rupture

The radiological consequences (control room and exclusion area boundary doses) of certain accidents increase from those of the existing analyses. However, in all cases, the results remain within the regulatory limits, as well as the regulatory guidance (i.e., Standard Review Plan criteria).

The NSSS systems, components, and supporting analyses, (e.g., structural integrity, performance capability) were verified to remain in compliance with the acceptance criteria associated with the existing licensing bases.

The BOP systems and components are capable of performing their design function for operation at the uprated conditions. The evaluation resulted in a revised calculation of main steam safety valve capacity which validated a total capacity greater that the total NSSS steam flow. The analyses concluded that reinforcement of the fifth point feedwater heater drain nozzles is required. This modification will be implemented prior to uprated operation.

The analyses and evaluations performed demonstrate that operation of Surry Units 1 and 2 at an uprated core power level of 2546 MWt, assuming implementation of the required physical modifications and Technical Specification changes, will meet regulatory acceptance criteria. Although uprating involves minor changes in certain plant parameters, the probability of accidents previously evaluated in the UFSAR will not be increased because these parameters remain within the design capabilities of plant systems and equipment. The possibility of an accident different from those already evaluated will not be created because uprating does not create any new accident precursors. The radiological consequences of certain accidents previously evaluated in the UFSAR will be increased, but the calculated doses remain within the regulatory limits. Therefore, operation at uprated power level prior to NRC review and approval would constitute an unreviewed safety question. The margin of safety as defined in the Technical Specifications will not be reduced by operation at the uprated power level because the evaluations performed demonstrate that the results continue to meet the applicable acceptance criteria, thereby preserving the existing margin of safety.

The Technical Specification changes that are administrative in nature do not impact the safety analyses in any way.

SIGNIFICANT HAZARDS CONSIDERATION

Virginia Electric and Power Company has reviewed the proposed License Amendment with regard to the criteria of 10CFR50.92 and has concluded that the License Amendment as proposed does not constitute a significant hazards consideration. The proposed changes revise the Surry Operating License, Technical Specifications, and Technical Specification Bases to accommodate plant operation at an uprated core power level of 2546 MWt. The supporting core uprating evaluations were performed in accordance with the methodology outlined in Westinghouse WCAP-10263, entitled "A Review Plan for Uprating the Licensed Power for a Pressurized Water Reactor." In addition, administrative changes are being made to the affected pages to eliminate the discussion of two loop operation and to capitalize defined words and system names. Specifically, operation of the Surry Power Station in accordance with the proposed changes will not:

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

The uprating involves minor changes in certain plant parameters (e.g., core power level, Reactor Coolant System average temperature, containment temperature limits, etc.). However, these parameter changes have been fully accounted for in revised accident analyses and remain within the design capabilities of plant systems and equipment. Since the changes do not significantly revise plant hardware or operating practices, no significant increase in the probability of a previously evaluated accident has been introduced.

The radiological consequences analysis associated with core uprating results in increased doses for certain events. Although this represents an increase in the consequences for such events, the calculated doses remain within the applicable regulatory limits and regulatory guidance. The largest increase in dose was less than 2.5 rem for the control room thyroid dose following the design bases LOCA. However, the calculated dose of 29 rem is within the GDC 19 limit. Therefore, we conclude no significant increase in the consequences of a previously evaluated accident has been introduced by the proposed core uprate.

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

The proposed changes do not introduce any significant changes in mode of operation or performance requirements, and, therefore, do not introduce any new failure modes or accident precursors. The minor changes in certain plant parameters have been accounted for in revised accident analyses and remain within the design capabilities of plant systems and equipment. Therefore, no possibility of a new or different kind of accident from any previously evaluated has been created.

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

The analyses performed to support uprated operation continue to meet the applicable acceptance criteria. These design limit values establish the margin of safety which is enveloped by the Technical Specifications. Operation at the proposed conditions, justified by the revised analyses, will not reduce this margin of safety.

Regarding increased radiological doses for certain events, criteria from applicable regulation and regulatory guidance continues to be met as discussed above. Therfore, the increase in doses is not considered a significant reduction in margin.

ATTACHMENT 3

SURRY POWER STATION

SURRY CORE UPRATE LICENSING REPORT