instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; Amdt. 42 10-17-78

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3114.4 megawatts thermal. Amdt. 237 5-22-03

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. , are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

D. (1) Deleted per Amdt. 82, 12-11-82.

(2) Secondary Water Chemistry Monitoring Amdt. 60 1-28-80

ENO shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. The program shall include:

(a) Identification of a sampling schedule for the critical parameters and control points for these parameters;

1.0 DEFINITIONS

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

1.1 a. RATED POWER

A steady state reactor thermal power of 3114.4 MWT.

b. THERMAL POWER

The total core heat transfer rate from the fuel to the coolant.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 Cold Shutdown Condition

When the reactor is subcritical by at least 1% $\Delta k/k$ and Tavg is $\leq$ 200°F*.

1.2.2 Hot Shutdown Condition

When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{avg} is > 200°F* and $\leq$ 555°F.

1.2.3 Reactor Critical

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

1.2.4 Power Operation Condition

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

* For the one time, fuel out, chemical decontamination program only, this value will be 250°F.

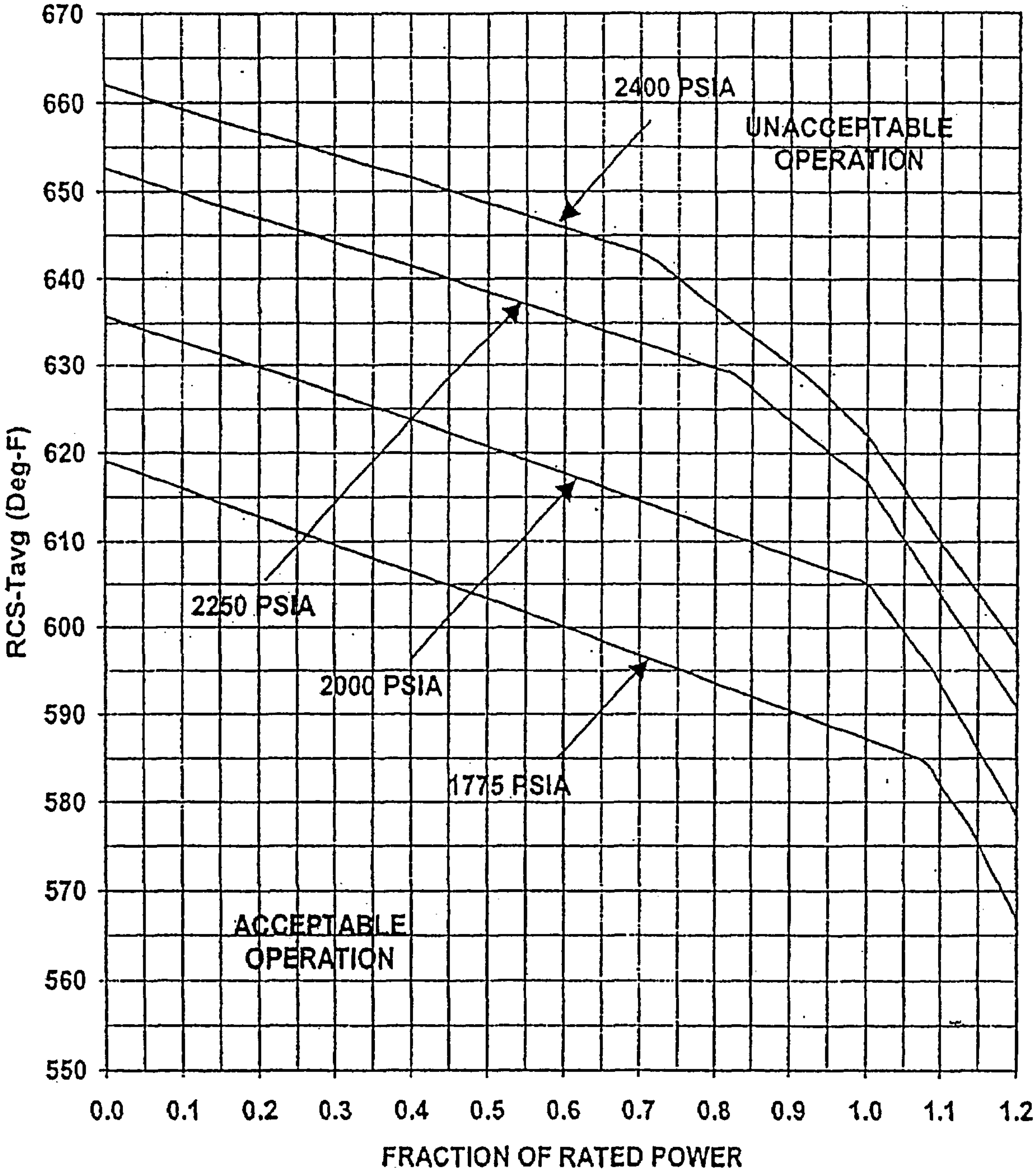

Figure 2.1-1

Reactor Core Safety Limit – Four Loops in Operation

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, and pressurizer level.

Objective

To provide for automatic protective action such that the principal process variables do not exceed a safety limit.

Specifications

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

      (1) High flux, power range (low setpoint): ≤ 25% of rated power.

   B. Core limit protection

      (1) High flux, power range (high setpoint): ≤ 109% of rated power.

      (2) High pressurizer pressure: ≤ 2363 psig.

      (3) Low pressurizer pressure: ≥ 1928 psig.

      (4) Overtemperature ΔT:

      $\Delta T \leq \Delta T_o \, [K_1 - K_2 \, (T - T') + K_3 \, (P - P') - f \, (\Delta I)]$

      where:

      ΔT = Measured ΔT by hot and cold leg RTDs, °F

      ΔT_o ≤ Indicated ΔT at rated power, °F

      T = Average temperature, °F

      T' = Design full power T_{avg} at rated power, ≤ 579.2°F

T'' = Indicated full power T_{avg} at rated power $\leq 579.2°F$

$K_4 \leq 1.074$

K_5 = Zero for decreasing average temperature

$K_5 \geq 0.188$, for increasing average temperature (sec/°F)

$K_6 \geq 0.0015$ for $T \geq T''$; $K_6 = 0$ for $T < T''$

$\frac{dT}{dt}$ = Rate of change of T_{avg}

(6) Low reactor coolant loop flow:

(a) ≥ 92% of normal indicated loop flow.

(b) Low reactor coolant pump frequency: ≥ 57.5 cps.

(7) Undervoltage: ≥ 70% of normal voltage.

C. Other reactor trips

(1) High pressurizer water level: ≤ 90% of span.

(2) Low-low steam generator water level: ≥ 7% of narrow range instrument span.

2. Protective instrumentation settings for reactor trip interlocks shall satisfy the following conditions:

A. The reactor trips on low pressurizer pressure, high pressurizer level, and low reactor coolant flow for two or more loops shall be unblocked when:

(1) Power range nuclear flux ≥ 10% of rated power, or

(2) Turbine first stage pressure ≥ 10% of equivalent full load.

B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates ≤ 60% of rated power.

G. REACTOR COOLANT SYSTEM PRESSURE, TEMPERATURE, AND FLOW RATE

Specifications

The following DNB related parameters pertain to four loop steady-state operation at power levels greater than 98% of rated full power:

a. Reactor Coolant System $T_{avg} \leq 586.7°F$

b. Pressurizer Pressure ≥ 2190 psia

c. Reactor Coolant System Total Flow Rate ≥ 331,840 gpm

Item (b), pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change in excess of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant temperature, T_{avg}, or pressurizer pressure exceed the values given in items (a) and (b), the parameter shall be restored to its applicable range within 2 hours.

Basis

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than the safety limit DNBRs.

The limits on reactor coolant system temperature, pressure and loop coolant flow represent those used in the accident analyses and are specified to assure that the values assumed in the accident analyses are not exceeded during steady-state four loop operation. Indicator uncertainties have not been accounted for in determining the DNB parameter limits on temperature and pressure.

If these requirements cannot be met, then:

1. maintain the plant in a safe, stable mode which minimizes the potential for a reactor trip, and

2. continue efforts to restore water supply to the auxiliary feedwater system, and

3. notify the NRC within 24 hours regarding the planned corrective action.

Basis

Reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system.

The operability of the twenty main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% Rated Thermal Power coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The total relieving capacity of the twenty main steam safety valves is 15,108,000 lbs/hr which is 111.2 percent of the total secondary steam flow of 13,580,000 lbs/hr at 100% NSSS Power (3126.4 Mwt). Startup and/or power operation is allowable with main steam safety valves inoperable within the limitations of Table 3.4-1 on the basis of the reduction in secondary system steam flow and thermal power required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are based on the heat removal capacity of the remaining operable steam line safety valves. The maximum thermal power corresponding to the heat removal capacity of the remaining operable steam line safety valves is determined via a conservative heat balance calculation as described in the attachment to Ref. 2 with an appropriate allowance for calorimetric power uncertainty.

TABLE 3.4-1

Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4-Loop Operation

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of Rated Thermal Power)
1	59
2	40
3	21

e. Control Bank Insertion Limits for Specification 3.10.4.

6.9.1.9 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specification 3.10.4 - Shutdown Bank Insertion Limit, Control Bank Insertion Limits and 3.10.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary). (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

c. T.M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

d. NUREG-0800, Standard Review Plan, US Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

e. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.10.2 Height Dependent Heat Flux Hot Channel Factor.)

f. WCAP-12945-P, Westinghouse "Code Qualification Document for Best Estimate LOCA Analyses", July, 1996

g. Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √™ System," Revision 0, March 1997, and Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM √™ System," Revision 0, May 2000.

6.9.1.10 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.