(4) FPL Energy Seabrook, LLC, pursuant to the Act and 10 CFR 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) FPL Energy Seabrook, LLC, pursuant to the Act and 10 CFR 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;

(6) FPL Energy Seabrook, LLC, pursuant to the Act and 10 CFR 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein; and

(7) DELETED

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and 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

FPL Energy Seabrook, LLC, is authorized to operate the facility at reactor core power levels not in excess of 3587 megawatts thermal (100% of rated power).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 101*, and the Environmental Protection Plan contained in Appendix B are incorporated into Facility License No. NPF-86. FPL Energy Seabrook, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) License Transfer to FPL Energy Seabrook, LLC

a. On the closing date(s) of the transfer of any ownership interests in Seabrook Station covered by the Order approving the transfer, FPL Energy Seabrook, LLC, shall obtain from each respective transferring owner all of the accumulated decommissioning trust funds for the facility, and ensure the deposit of such funds and additional funds, if necessary, into a decommissioning trust or trusts for Seabrook Station established by FPL Energy Seabrook, LLC, such that the amount of such funds deposited meets or exceeds the amount required under 10 CFR 50.75 with respect to the interest in Seabrook Station FPL Energy Seabrook, LLC, acquires on such dates(s).

* Implemented

J. <u>Additional Conditions</u>

The Additional Conditions contained in Appendix C, as revised through Amendment No. 94, are hereby incorporated into this license. FPL Energy Seabrook, LLC, shall operate the facility in accordance with the Additional Conditions.

K. <u>Inadvertent Actuation of the Emergency Core Cooling System (ECCS)</u>

Prior to startup from refueling outage 11, FPL Energy Seabrook commits to either upgrade the controls for the pressurizer power operated relief valves (PORV) to safety-grade status and confirm the safety-grade status and water-qualified capability of the PORVs, PORV block valves and associated piping or to provide a reanalysis of the inadvertent safety injection event, using NRC approved methodologies, that concludes that the pressurizer does not become water solid within the minimum allowable time for operators to terminate the event.

3. This License is effective as of the date of issuance and shall expire at midnight on October 17, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION

(Original signed by:
Thomas E. Murley)

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Appendix A - Technical Specifications (NUREG-1386)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional Conditions

Date of Issuance: March 15, 1990

PHYSICS TESTS

1.23 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.24 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3587 Mwt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.29 The RTS RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its RTS Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS (SLs)

2.1.1 REACTOR CORE SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.14 for the WRB-2M DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained less than or equal to 2735 psig.

2.1.3 SAFETY LIMIT VIOLATIONS

2.1.3.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.3.2 If SL 2.1.2 is violated:

a. In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

b. In MODE 3, 4, or 5, restore compliance within 5 minutes.

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. Pressurizer Water Level - High	8.0	4.20	0.84	≤92% of instrument span	≤93.75% of instrument span
12. Reactor Coolant Flow - Low	N.A.	N.A.	N.A.	≥90% of indicated loop flow	≥89.6% of indicated loop flow
13. Steam Generator Water Level Low - Low	N.A.	N.A.	N.A.	≥20.0% of narrow range instrument span	≥19.5% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	15.0	1.39	0	≥10,200 volts	≥9,822 volts
15. Underfrequency - Reactor Coolant Pumps	2.9	0	0	≥55.5 Hz	≥55.3 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥500 psig	≥450 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{(1)}{(1+\tau_3 S)} \leq \Delta T_0 \{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} [T \frac{(1)}{(1+\tau_6 S)} - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured RCS ΔT by RTD Instrumentation, °F;

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, values specified in the COLR;

$\frac{1}{1+\tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, value specified in the COLR;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER, °F;

K_1 = Value specified in the COLR;

K_2 = Value specified in the COLR;

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in lead-lag compensator for T_{avg}, values specified in the COLR;

T = Measured RCS Average temperature, °F;

$\frac{1}{1+\tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, value specified in the COLR;

TABLE NOTATIONS

NOTE 1: (Continued)

T'		Indicated RCS T_{avg} at RATED THERMAL POWER, °F, (Calibration temperature for ΔT instrumentation, value specified in the COLR);
K_3	=	Value specified in COLR;
P	=	Measured Pressurizer pressure, psig;
P'	=	Nominal RCS operating pressure, psig, value specified in the COLR;
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers as specified in the COLR.

NOTE 2: Cycle dependent values for the channel's Allowable Value are specified in the COLR.

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{(1)}{(1+\tau_3 S)} \le \Delta T_0 \{K_4 - K_5 \frac{(\tau_7 S)}{(1+\tau_7 S)} \frac{(1)}{(1+\tau_6 S)} T - K_6 [T \frac{(1)}{(1+\tau_6 S)} - T''] - f_2(\Delta I)\}$$

Where: ΔT = As defined in Note 1,

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1+\tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 = Value specified in the COLR,

K_5 = Value specified in the COLR,

$\frac{\tau_7 S}{1+\tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in rate-lag compensator for T_{avg}, value specified in the COLR,

$\frac{1}{1+\tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	Value specified in COLR,
T	=	As defined in Note 1,
T"	=	Indicated T_{avg} at RATED THERMAL POWER, °F, (Calibration temperature for ΔT instrumentation, value specified in the COLR),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	A function of the indicated difference between the top and bottom detectors of the power-range neutron ion chambers as specified in the COLR.

NOTE 4: Cycle dependent values for the channel's Allowable Value are specified in the COLR.

g. The limits specified in Specification 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:

1) Lower core region from 0 to 10%, inclusive.

2) Upper core region from 90 to 100%, inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% when using the moveable incore detectors or 5.21% when using the fixed incore detectors to account for measurement uncertainty.

4.2.2.4 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg} is less than or equal to the limit specified in the COLR,

b. Pressurizer Pressure is greater than or equal to the limit specified in the COLR*, and

c. Reactor Coolant System Flow shall be:

1. $\geq$ 374,400 gpm**; and,

2. $\geq$ 383,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by an approved method to be within its limit prior to operation above 95% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**Thermal Design Flow. An allowance for measurement uncertainty shall be made when comparing measured flow to Thermal Design Flow.

***Minimum measured flow used in the Revised Thermal Design Procedure.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4.	Steam Line Isolation					
a.	Manual Initiation (System)	N.A.	N.A.	N.A.	N.A.	N.A.
b.	Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c.	Containment Pressure--Hi-2	5.2	0.71	1.67	≤4.3 psig	≤5.3 psig
d.	Steam Line Pressure--Low	13.1	10.71	1.63	≥585 psig	≥568 psig*
e.	Steam Generator Pressure - Negative Rate--High	3.0	0.5	0	≤100 psi	≤123 psi**
5.	Turbine Trip					
a.	Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b.	Steam Generator Water Level--High-High (P-14)	N.A	N.A.	N.A.	≤90.8% of narrow range instrument span.	≤91.3% of narrow range instrument span.
6.	Feedwater Isolation					
a.	Steam Generator Water Level--Hi-Hi-(P-14)	N.A.	N.A.	N.A.	≤90.8% of narrow range instrument span.	≤91.3% of narrow range instrument span.
b.	Safety Injection	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Emergency Feedwater					
a. Manual Initiation					
(1) Motor driven pump	N.A.	N.A.	N.A.	N.A.	N.A.
(2) Turbine driven pump	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low Start Motor-Driven Pump and Start Turbine-Driven Pump	N.A.	N.A.	N.A.	≥20.0% of narrow range instrument span.	≥19.5% of narrow range instrument span.
d. Safety Injection Start Motor-Driven Pump and Turbine-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9. for Loss-of-Offsite Power Setpoints and Allowable Values.				
8. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level –Low-Low	4.0*** 2.1****	1.0	2.8	120,478 gals.	≤121,521*** gals. ≥119,435**** gals.
Coincident With Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR-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	60
2	42
3	25

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					
Loop 1	Loop 2	Loop 3	Loop 4	LIFT SETTING* (± 3%)**	ORIFICE SIZE
V6	V22	V36	V50	1185 psig	16.0 sq. in.
V7	V23	V37	V51	1195 psig	16.0 sq. in.
V8	V24	V38	V52	1205 psig	16.0 sq. in.
V9	V25	V39	V53	1215 psig	16.0 sq. in.
V10	V26	V40	V54	1225 psig	16.0 sq. in.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**Within ±1% following main steam line Code safety valve testing.

6.8.1.6.a. (Continued)

12. Cycle dependent DNB-related parameters for reactor coolant system average temperature (T_{avg}), and pressurizer pressure for Specification 3.2.5.

13. The boron concentration limits for MODES 1, 2 and 3 for Specification 3.5.1.1.

14. The boron concentration limits for MODES 1, 2, 3 and 4 for Specification 3.5.4.

15. The boron concentration limits for MODE 6 for Specification 3.9.1.

6.8.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-12945-P-A, "Code Qualification Document for Best Estimate LOCA Analysis," Volume 1, Revision 2, and Volumes 2 through 5, Revision 1; Bajorek, S. M., et al, 1998.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

2. WCAP-10079-P-A, (Proprietary) and WCAP-10080-A (Nonproprietary), "NOTRUMP: A Nodal Transient Small Break and General Network Code", August 1985.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

3. YAEC-1363-A, "CASMO-3G Validation," April, 1988.

   YAEC-1659-A, "SIMULATE-3 Validation and Verification," September, 1988.

   WCAP-11596-P-A, (Proprietary), "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores", June, 1988.

   WCAP-10965-P-A, (Proprietary), "ANC: A Westinghouse Advanced Nodal Computer Code", September, 1986.

6.8.1.6.b. (Continued)

Methodology for Specifications:
3.1.1.1 - SHUTDOWN MARGIN for MODES 1,2, 3, and 4
3.1.1.2 - SHUTDOWN MARGIN for MODE 5
3.1.1.3 - Moderator Temperature Coefficient
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

4. Seabrook Station Updated Final Safety Analysis Report, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant System".

Methodology for Specifications:
3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
3.1.1.2 - SHUTDOWN MARGIN for MODE 5

5. YAEC-1241, "Thermal-Hydraulic Analysis of PWR Fuel Elements Using the CHIC-KIN Code", R. E. Helfrich, March, 1981.

WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", October, 1999.

WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

Methodology for Specification:
2.1 - Safety Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor
3.2.5 - DNB Parameters

6. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 For PWR Applications," October, 1992.

WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure", April, 1989.

WCAP-8745-P-A, "Design Basis for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September, 1986.

6.8.1.6.b. (Continued)

Methodology for Specification:
2.2.1 - Limiting Safety System Settings
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

7. YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station," October, 1992

Methodology for Specification:
2.2.1 - Limiting Safety System Settings
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

8. YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications," December, 1992.

Methodology for Specification:
2.2.1 - Limiting Safety System Settings
3.1.1.3 - Moderator Temperature Coefficient
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

9. YAEC-1752, "STAR Methodology Application for PWRs, Control Rod Ejection, Main Steam Line Break," October, 1990.

Methodology for Specification:
3.1.1.3 - Moderator Temperature Coefficient
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits
3.2.1 - AXIAL FLUX DIFFERENCE
3.2.2 - Heat Flux Hot Channel Factor
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b. (Continued)

10. YAEC-1855PA, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis," October, 1992.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

11. YAEC-1624P, "Maine Yankee RPS Setpoint Methodology Using Statistical Combination of Uncertainties - Volume 1 - Prevention of Fuel Centerline Melt," March, 1988.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

12. NYN-95048, Letter from T. C. Feigenbaum (NAESCo) to NRC, "License Amendment Request 95-05: Positive Moderator Temperature Coefficient", May 30, 1995.

    Methodology for Specification:
    3.1.1.3 - Moderator Temperature Coefficient

13. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report". April, 1995, (Westinghouse Proprietary).

    Methodology for Specification:
    3.2.2 - Heat Flux Hot Channel Factor

14. WCAP-10216-P-A, Revision 1A (Proprietary), "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification", February, 1994.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. There is additional margin available to offset any other DNBR penalties and for plant design flexibility.

When an $F_Q(Z)$ measurement is taken, an allowance for both measurement error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the movable incore detectors, while 5.21% is appropriate for surveillance results determined with the fixed incore detectors. A 3% allowance is appropriate for manufacturing tolerance.

The hot channel factor $F^M_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to Relaxed Axial Offset Control (RAOC) operation, W(Z), to provide assurance that the limit on the hot channel factor $F_Q(Z)$ is met. W(Z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(Z) function for normal operation is specified in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

When RCS $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the established limit. Appropriate $F^N_{\Delta H}$ measurement uncertainties are already incorporated into the limits $F^N_{\Delta H}$ established in the CORE OPERATING LIMITS REPORT for each measurement system, and a bounding $F^N_{\Delta H}$ measurement uncertainty has been applied in determination of the design DNBR value. The appropriate $F^N_{\Delta H}$ measurement uncertainties are 4.13% for the fixed incore detector system and 4% for the movable incore detector system.

3/4.2.4 QUADRANT POWER TILT RATIO

The purpose of this specification is to detect gross changes in core power distribution between monthly Incore Detector System surveillances. During normal operation the QUADRANT POWER TILT RATIO is set equal to 1.0 once acceptability of core peaking factors has been established by review of incore surveillances. The limit of 1.02 is established as an indication that the power distribution has changed enough to warrant further investigation.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia 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 specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, (1974 Edition, including the Summer 1975 Addenda). The total relieving capacity for all valves on all of the steam lines is 1.816×10^7 lbs/hr which is 109.8% of the total secondary steam flow of 1.654×10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant 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 derived on the following bases:

For four loop operations:

$$\text{Hi } \varnothing = (100/Q_{rated}) \times \left[\frac{(W_s \times h_{fg} \times N)}{K} - Q_{rcp}\right]$$

where:

Hi Ø	=	Safety Analysis power range high neutron flux setpoint, percent of RATED THERMAL POWER
Q_{rated}	=	RATED THERMAL POWER, Mwt
Q_{rcp}	=	Reactor coolant pump heat, Mwt
K	=	Conversion factor, 3.412×10^6 (Btu/hr)/Mwt
h_{fg}	=	heat of vaporization for steam at 110% of the Secondary System design pressure, Btu/lbm
N	=	Number of loops in plant
W_s	=	Minimum total steam flow rate, lbm/hr, of the operable MSSVs on any one steam generator at the MSSV inlet pressure which assures all Secondary System pressures are no greater than 110% of design.