



FPL Energy
Seabrook Station

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SEP 13 2004

Docket No. 50-443
SBK-L-04044

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Seabrook Station
Facility Operating License NPF-86
Changes to License Amendment Request 04-03,
Application for Stretch Power Uprate

References:

- 1 FPL Energy Seabrook, LLC, letter NYN-04016, LAR 04-03, "Application for Stretch Power Uprate," dated March 17, 2004.
- 2 FPL Energy Seabrook, LLC, letter NYN-02022, LAR 01-10, "Relocation of Cycle Specific Parameters to the Core Operating Limits Report," dated April 15, 2002.
- 3 Seabrook Station, Unit No. 1, License Amendment 96 - Issuance of Amendment Re: "Relocation of Cycle-Specific Parameters to the Core Operating Limits Report (TAC NO. MB4918)", dated March 23, 2004.

By letter dated March 17, 2004 (Reference 1), FPL Energy Seabrook, LLC requested an amendment to facility operating license NPF-86 and the plant technical specifications for Seabrook Station. This license amendment request (LAR) is an application for a stretch power uprate (SPU) which will increase the Seabrook Station licensed reactor core power by 5.2 percent from 3411 megawatts thermal (MWt) to 3587 MWt.

FPL Energy Seabrook, LLC has since identified analysis errors in the LAR. To correct the LAR, FPL Energy Seabrook, LLC is submitting the following changes:

- LAR Attachment 1, Subsection 6.3.5.5.2, Method of Analysis – Dilution During Refueling, the last bullet in this subsection (top of page 6-193), is revised from 2000 ppm to 2100 ppm, such that it reads, "The initial boron concentration is assumed to be 2100 ppm..."

A001

- LAR Attachment 1 Subsection 6.3.5.5.3, Results – Dilution During Refueling (page 6-195), is revised from 37.14 minutes to 33.47 minutes, such that it reads, “For dilution during refueling, the minimum time required for the shutdown margin to be lost is 33.47 minutes.”
- LAR Attachment 1 Subsection 6.3.6, Increase In Reactor Coolant Inventory
 - LAR Subsections 6.3.6.1.2, 6.3.6.1.3, and 6.3.6.1.4, (pages 6-235 and 6-236) – The operator action time in the last paragraph of each subsection is revised from 10 minutes to 10.1 minutes.
 - LAR Table 6.3.6.1-1 (page 6-239) - the Time entries for both the “Operator Stops Safety Injection” and “Maximum Pressurizer Pressure” are revised to 606.01 seconds and the Maximum Pressurizer Pressure reached is being revised from 2378.18 psia to 2386.19 psia.
 - LAR Figures 6.3.6.1-1, 6.3.6.1-2, and 6.3.6.1-3 (pages 6-241 – 6-243, respectively) are replaced with the figures contained in Enclosure 3 to this letter.

The following revisions correct an incorrect change description in LAR 04-03.

- LAR Subsections 6.1.3.1, Post-LOCA Subcriticality and Long-Term Core Cooling – Introduction and Background, 6.1.4.2, Containment Sump pH Control – Evaluation, and 6.1.5.1, Hot Leg Switchover – Introduction and Background (pages 6-8, 6-10, and 6-11, respectively) incorrectly state that a proposed technical specification change to reduce the refueling water storage tank and accumulator boron concentration in TS 3/4.5.1 and 3/4.5.4, respectively, was included in the LAR. These TS changes were to reflect the safety analyses performed to support the Seabrook Station LAR using the reduced refueling water storage tank and accumulator boron concentrations. However, the LAR should have stated that these proposed TS changes were not submitted as part of LAR 04-03 because a previous submittal (LAR 01-10, Ref. 2) was pending NRC approval to relocate the minimum and maximum required boron concentrations for the refueling water storage tank and accumulators to the Core Operating Limits Report (COLR). LAR 01-10 was subsequently approved (Ref. 3) following the submittal of LAR 04-03. Therefore, to correct the incorrect reference to TS 3/4.5.1 and 3/4.5.4 the following changes have been made:
 - LAR Subsection 6.1.3.1 (page 6-8) – The first paragraph is revised to delete the first two sentences and replaced with the following, “The minimum refueling water storage tank and accumulator boron concentrations used in the safety analyses for the SPU were reduced from 2700 ppm to 2400 ppm and 2600 ppm to 2300 ppm, respectively.”
 - LAR Subsection 6.1.4.2 (page 6-10) – The first paragraph is revised to delete the last sentence beginning, “However, as indicated...” The second paragraph first sentence is deleted and replaced with the following: “The containment spray and sump pH were evaluated using boron concentration ranges for the refueling water storage tank and accumulators of 2400 to 2600 ppm and 2300 to 2600 ppm, respectively.”

- LAR Subsection 6.1.5.1 (page 6-11) – the first paragraph, first sentence is deleted and replaced with the following: “The maximum refueling water storage tank and accumulator boron concentration used in the safety analyses for the SPU was 2600 ppm.”

In addition to the errors discussed above, FPL Energy Seabrook has identified several typographical errors in the LAR and is submitting the following changes:

- LAR Attachment 1 – Table 4.3-2 (page 4-31), “Seabrook Station SPU Summary of Reactor Trip System / Engineered Safety Feature Actuation System Setpoint Calculations,”
 - Overtemperature ΔT Reactor Trip $f(\Delta I)$ function incorrectly states “between (-30% and +8%).” The correct range is “-20% and +8%.”
 - Overtemperature ΔT Reactor Trip – Negative Slope incorrectly states “ $\Delta I < -30\%$.” The correct value is “ $\Delta I < -20\%$.”
- LAR Attachment 1 Subsection 6.4.5.2 (page 6-282), in the second paragraph on this page, beginning, “The highest peak pressure under SPU conditions...,” an editing error resulted in the inclusion of an incorrect statement. This paragraph is being revised to delete the following sentence in its entirety, “The peak pressure for a main steamline break has changed from a double-ended rupture at near zero power (hot standby) to the double-ended rupture at 100% power under SPU conditions.”
- LAR Attachments 2 and 3, page 3 of the Facility Operating License – FPLE Seabrook was changed to FPL Energy Seabrook in the original submittal, and LLC has been added for further consistency.
- LAR Attachments 2 and 3, Technical Specification 2.1.1.1 (page 2-1), corrected typographical error “ration” to “ratio.” The original Technical Specification had the correct spelling.
- LAR Attachments 2 and 3, Technical Specification 2.1.1.2. (page 2-1), corrected typographical error “5090°F” to “5080°F.” The original Technical Specification had the correct value.
- LAR Attachments 2 and 3, page 2-5 in both attachments, the table is incorrectly numbered as “Table 2.1-1.” The correct table number is “Table 2.2-1.”
- LAR Attachments 2 and 3, pages 2-7 and 2-9 showing editorial changes were inadvertently omitted.
- LAR Attachments 2 and 3, Technical Specification 3/4.2.2 (page 3/4 2-6b), corrected editorial error “Limiting Condition For Operation” to “Surveillance Requirements.” The original Technical Specification had the correct header title.

- LAR Attachment 3, Technical Specification 3.2.5c (page 3/4 2-10), corrected RCS flow values from “374,000 gpm” and “383,000 gpm” to “374,400 gpm” and “383,800 gpm,” respectively, to reflect the markup version of the Technical Specification in Attachment 2.
- LAR Attachments 2 and 3, Technical Specification 3.2.5 *footnote (page 3/4 2-10), corrected typographical error “or” to “of.” The original Technical Specification had the correct spelling.
- LAR Attachment 2, Table 3.3-4, Functional Unit 5.b and 6.a, Trip Setpoint (page 3/4 3-26), corrected typographical error “86.8%” to “86.0%.” The original Technical Specification had the correct value.
- LAR Attachments 2 and 3, Table 3.3-4, the setpoint and allowable value for Functional Unit 7.c. are incorrectly labeled as “less than or equal to (\leq).” The setpoint and allowable value should be “greater than or equal to (\geq)” as indicated in the original Technical Specifications.
- LAR Attachments 2 and 3, Table 3.3-4, Functional Unit 8.b, Trip Setpoint (page 3/4 3-27), corrected typographical error “120.478 gals.” to “120,478 gals.” The original Technical Specification had the correct value.
- LAR Attachments 2 and 3, Technical Specification Bases 3/4.7 (page B 3/4 7-1), corrected editorial error “Turbine Systems” to “Plant Systems.” The original Technical Specification had the correct header title.
- LAR Attachment 6, Sections 1.3 and 1.4 (pages 4 and 5 respectively), the table is incorrectly referenced as “Table 2.1-1.” The correct table reference is “Table 2.2-1.”
- LAR Attachment 6, Sections 1-5 (page 6), second bullet, the specification reference should be 4.2.2.2.g.2 in lieu of 4.2.2.2.g.1.
- LAR Attachment 6, Section 1.7 (page 9), the setpoints and allowable values are incorrectly labeled as “greater than or equal to (\geq).” The setpoints and allowable values in this section should all be “less than or equal to (\leq)” as indicated in LAR Attachments 2 and 3 on Table 3.3-4. Also, the current trip setpoint is 86.0%, not 86.8% as indicated in the text.
- LAR Attachment 6, Section 1.10, first bullet at the top of page 14 - the paragraph should be deleted and replaced with the following; “The following paragraph is added, “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. ” ”

Enclosures 1 and 2 to this letter contain complete revised markup and retyped Technical Specification pages for LAR 04-03, Attachments 2 and 3 incorporating changes as a result of the above. Additionally, subsequent to submittal of LAR 04-03, LAR 01-10 was approved by the NRC as licensed amendment 96 (Ref. 3) and is incorporated in this submittal.

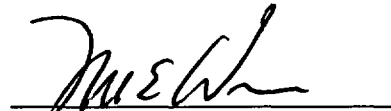
The changes described herein do not change FPL Energy Seabrook's conclusion that the proposed license amendment request, LAR 04-03, does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment continues to meet the categorical exclusion requirements of 10 CFR 51.22(c)(9) and an environmental impact appraisal need not be prepared.

The analysis and typographical errors have been entered into the FPL Energy Seabrook Corrective Action Program and corrective actions have been taken. The analysis errors were determined to only affect the referenced section of the LAR. The typographical errors were the result of process deficiencies that have been corrected.

Should you have any questions concerning this information, please contact Mr. Stephen T. Hale, Power Uprate Project Manager, at (603) 773-7561.

Very truly yours,

FPL Energy Seabrook, LLC

A handwritten signature in black ink, appearing to read 'Mark E. Warner', is written over a horizontal line.

Mark E. Warner
Site Vice President

cc. S. J. Collins, NRC Region I Administrator
S. P. Wall, NRC Project Manager, Project Directorate I-2
G. T. Dentel, NRC Resident Inspector

Mr. Bruce Cheney, Director
New Hampshire Bureau of Emergency Management
State Office Park South
107 Pleasant Street
Concord, NH 03301

Oath and Affirmation

I, Mark E. Warner, Site Vice President of FPL Energy Seabrook, LLC hereby affirm that the information and statements contained within this change to License Amendment Request 04-03 are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

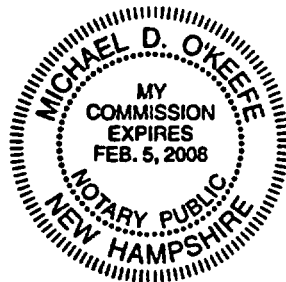
Sworn and subscribed
before me this

13 day of September, 2004

Michael O'Keefe
Notary Public



Mark E. Warner
Site Vice President



ENCLOSURE 1 TO SBK-L-04044

REVISED MARKUP TECHNICAL SPECIFICATION PAGES

**ATTACHMENT 2
MARKUP OF PROPOSED CHANGES TO THE
FACILITY OPERATING LICENSE AND
TECHNICAL SPECIFICATIONS**

LIST OF PAGES CONTAINING PROPOSED CHANGES

Page Number	Current Amendment Number
Facility Operating License	
3	86
Technical Specifications	
1-5	81
2-1	96
2-5	77
2-7	76
2-8	76
2-9	76
2-10	96
3/4 2-6b	76
3/4 2-10	96
3/4 3-26	45
3/4 3-27	60
3/4 7-2	43
6-18B	96
6-18C	76
6-18D	76
Technical Specification Bases	
B 3/4 2-3	76
B 3/4 7-1	43

The attached markup pages reflect the currently issued revision of the Facility Operating License NPF-86 and the Seabrook Station Technical Specifications and Bases. Pending changes are not reflected in the attached markup pages.

- (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 ~~3411~~ megawatts thermal (100% of rated power).

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(2) Technical Specifications

FPL Energy Seabrook, LLC,

The Technical Specifications contained in Appendix A, as revised through Amendment No. *, 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

Amendment No. 86,

DEFINITIONS

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

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1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3444 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 ~~1.17~~ for the ~~WRB-1/WRB-2/~~ WRB-2M DNB correlations.

1.14

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

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
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	44.0	12.53	0.55	≥44.0% of narrow range instrument span	≥42.6% 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.

N.A


20.0

19.5

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \frac{(1)}{(1 + \tau_6 S)} - T^1] + K_3(P - P^1) - f_1(\Delta I)\}$$


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 2.2-1 (Continued)

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.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_0 \{K_4 - K_5 \frac{(\tau_7 S)}{(1 + \tau_7 S)} \frac{(1)}{(1 + \tau_6 S)} T - K_6 [\overset{\boxed{..}}{\frac{(1)}{(1 + \tau_6 S)} - T^{+4}}] - f_2(\Delta I)\}$$

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 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = Value specified in COLR,

T = As defined in Note 1,

T'' → 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.

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

SURVEILLANCE REQUIREMENTS

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 ~~15%~~, inclusive.

10

2) Upper core region from ~~85~~ to 100%, inclusive.

90

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)

POWER DISTRIBUTION LIMITS

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) $\geq 382,800$ gpm**, and
 - 2) $\geq 392,800$ gpm***

374,400

383,800

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 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

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
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)	4.0 <div style="border: 1px solid black; padding: 2px; width: fit-content; margin: 0 auto;">N.A.</div>	2.24 <div style="border: 1px solid black; padding: 2px; width: fit-content; margin: 0 auto;">N.A.</div>	0.55 <div style="border: 1px solid black; padding: 2px; width: fit-content; margin: 0 auto;">N.A.</div>	≤86.0% of narrow range instrument span. <div style="border: 1px solid black; padding: 2px; width: fit-content; margin: 0 auto;">90.8</div>	≤87.7% of narrow range instrument span. <div style="border: 1px solid black; padding: 2px; width: fit-content; margin: 0 auto;">91.3</div>
6. Feedwater Isolation					
a. Steam Generator Water Level--Hi-Hi-(P-14)	4.0	2.24	0.55	≤86.0% of narrow range instrument span.	≤87.7% 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

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
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	14.0 ↑ N.A.	12.53 ↑ N.A.	0.55 ↑ N.A.	≥14.0% of narrow range instrument span. ↑ 20.0	≥12.6% of narrow range instrument span. ↑ 19.5
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 Coincident With Safety Injection	4.0*** 2.1****	1.0	2.8	120,478 gals.	≤121,521*** gals. ≥119,435**** gals.
	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

<u>MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR</u>	<u>MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)</u>
--	--

1
2
3

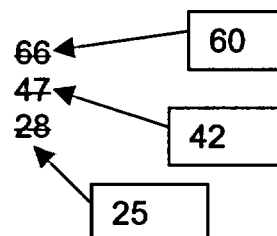


TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER

<u>Loop 1</u>	<u>Loop 2</u>	<u>Loop 3</u>	<u>Loop 4</u>	<u>LIFT SETTING* (± 3%)**</u>	<u>ORIFICE SIZE</u>
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.

ADMINISTRATIVE CONTROLS

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.

-A → ~~WCAP-14565-P~~, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", April, 1997. ← October, 1999.

~~Letter from T. H. Essig (NRC) to H. Sepp (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-14565-P, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", January, 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.

ADMINISTRATIVE CONTROLS

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

~~WCAP-14551-P, (Proprietary), "Westinghouse Setpoint Methodology for Protection Systems, Seabrook Nuclear Power Station Unit 1, 24 Month Fuel Cycle Evaluation", June, 1998.~~

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

ADMINISTRATIVE CONTROLS

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.

~~WCAP-8385-P, (Proprietary), "Power Distribution Control and Load Following Procedures", September, 1974.~~

Methodology for Specification:

- 3.2.1 - AXIAL FLUX DIFFERENCE
- 3.2.2 - Heat Flux Hot Channel Factor

POWER DISTRIBUTION LIMITS

There is additional margin available to offset any other DNBR penalties and

BASES

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

$F_{\Delta H}^N$ will be maintained within its limits provided Conditions a. through d. above are maintained. ~~The design limit DNBR includes margin to offset any rod bow penalty. Margin is also maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is available 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 $F_Q^M(Z)$ 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 $F_{\Delta H}^N$ is measured, no additional allowances are necessary prior to comparison with the established limit. ~~A bounding measurement error of 4.13% for $F_{\Delta H}^N$ has been allowed for in determination of the design DNBR value.~~

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.

Appropriate $F_{\Delta H}^N$ measurement uncertainties are already incorporated into the $F_{\Delta H}^N$ limits established in the CORE OPERATING LIMITS REPORT for each measurement system, and a bounding $F_{\Delta H}^N$ measurement uncertainty has been applied in determination of the design DNBR value. The appropriate $F_{\Delta H}^N$ measurement uncertainties are 4.13% for the fixed incore detector system and 4% for the movable incore detector system.

3/4.7 PLANT SYSTEMS

BASES

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).

109.8

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 120% of the total secondary steam flow of 1.514×10^7 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.

1.654

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:

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

where:

- Hi \emptyset = 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.

ENCLOSURE 2 TO SBK-L-04044

REVISED RETYPED TECHNICAL SPECIFICATION PAGES

**ATTACHMENT 3
RETYPE PAGES FOR PROPOSED CHANGES TO THE
FACILITY OPERATING LICENSE AND
TECHNICAL SPECIFICATIONS**

LIST OF PAGES CONTAINING PROPOSED CHANGES

Page Number	Current Amendment Number
Facility Operating License	
3	86
Technical Specifications	
1-5	81
2-1	96
2-5	77
2-7	76
2-8	76
2-9	76
2-10	76
3/4 2-6b	76
3/4 2-10	96
3/4 3-26	45
3/4 3-27	60
3/4 7-2	43
6-18B	96
6-18C	76
6-18D	76
Technical Specification Bases	
B 3/4 2-3	76
B 3/4 7-1	43

The attached retyped pages reflect the currently issued revision of the Facility Operating License NPF-86 and the Seabrook Station Technical Specifications and Bases. Pending changes are not reflected in the attached retyped pages.

- (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. * , 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).

DEFINITIONS

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

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
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

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

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 2.2-1 (Continued)

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);

K₃ = 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.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where: ΔT = As defined in Note 1,

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

τ_1, τ_2 = As defined in Note 1,

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

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,

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

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

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

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)

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.

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

SURVEILLANCE REQUIREMENTS

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)

POWER DISTRIBUTION LIMITS

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 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

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
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

<u>FUNCTIONAL UNIT</u>	<u>TOTAL ALLOWANCE (TA)</u>	<u>Z</u>	<u>SENSOR ERROR (S)</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
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 Coincident With Safety Injection	4.0*** 2.1****	1.0	2.8	120,478 gals.	≤121,521*** gals. ≥119,435**** gals.
	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

<u>MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR</u>	<u>MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)</u>
1	60
2	42
3	25

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER

<u>Loop 1</u>	<u>Loop 2</u>	<u>Loop 3</u>	<u>Loop 4</u>	<u>LIFT SETTING* ($\pm 3\%$)**</u>	<u>ORIFICE SIZE</u>
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 $\pm 1\%$ following main steam line Code safety valve testing.

ADMINISTRATIVE CONTROLS

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.

ADMINISTRATIVE CONTROLS

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

ADMINISTRATIVE CONTROLS

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

POWER DISTRIBUTION LIMITS

BASES

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

$F_{\Delta H}^N$ 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 $F_Q^M(Z)$ 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 $F_{\Delta H}^N$ is measured, no additional allowances are necessary prior to comparison with the established limit. Appropriate $F_{\Delta H}^N$ measurement uncertainties are already incorporated into the limits $F_{\Delta H}^N$ established in the CORE OPERATING LIMITS REPORT for each measurement system, and a bounding $F_{\Delta H}^N$ measurement uncertainty has been applied in determination of the design DNBR value. The appropriate $F_{\Delta H}^N$ 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 PLANT SYSTEMS

BASES

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:

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

where:

- Hi \emptyset = 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.

ENCLOSURE 3 TO SBK-L-04044

**LAR-04-03, SUBSECTION 6.3.6
REVISED FIGURES**

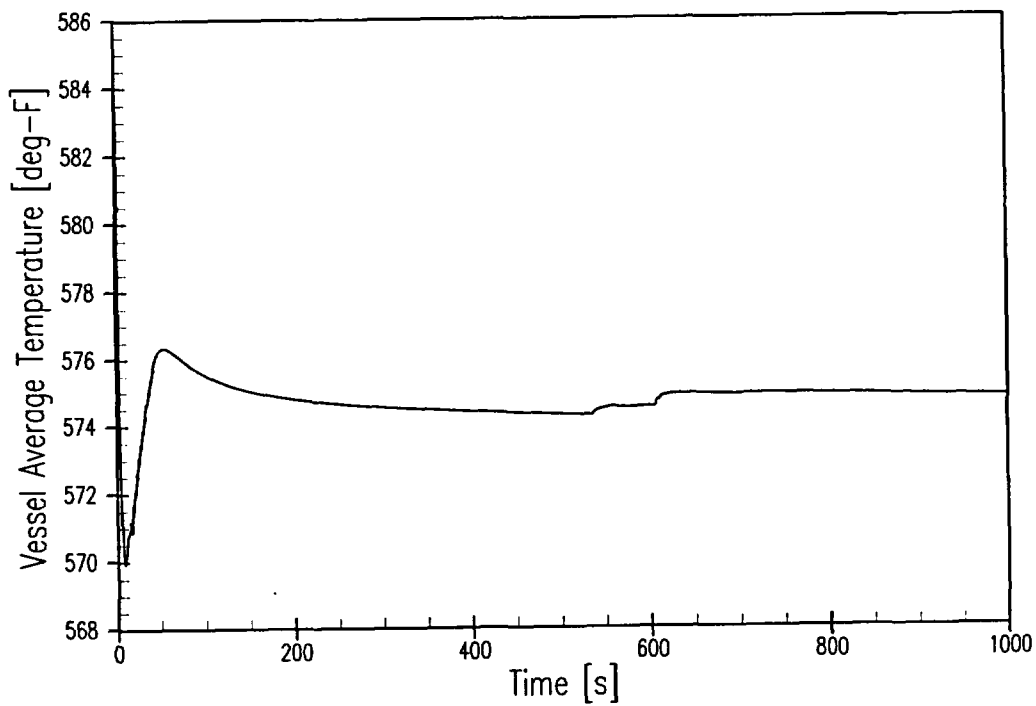
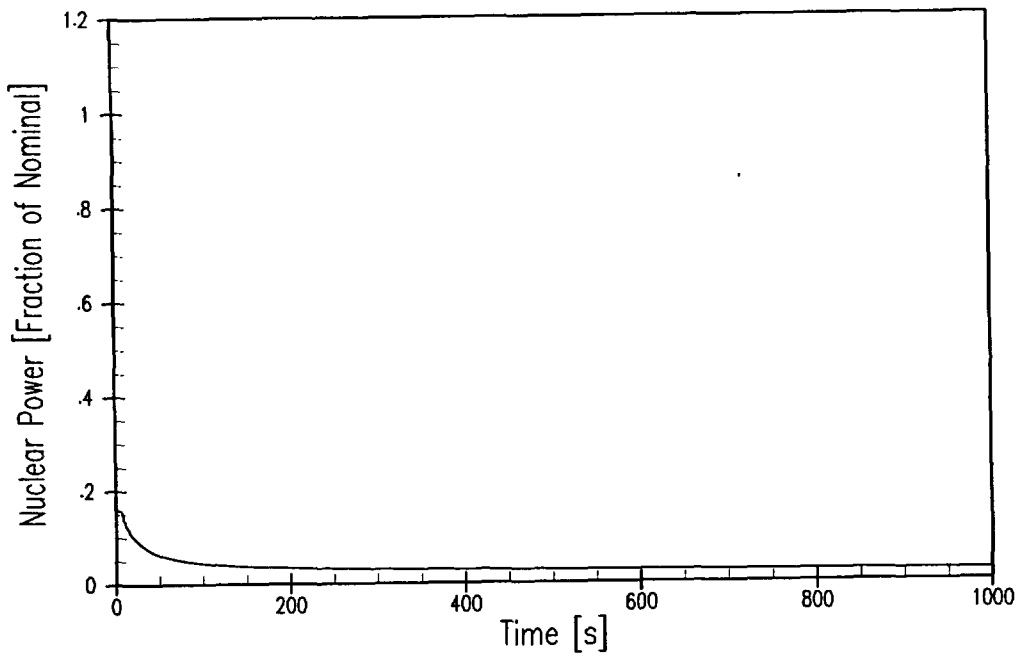


Figure 6.3.6.1-1
Inadvertent Emergency Core Cooling Systems Operation
During Power Operation
Nuclear Power and Vessel Average Temperature versus Time

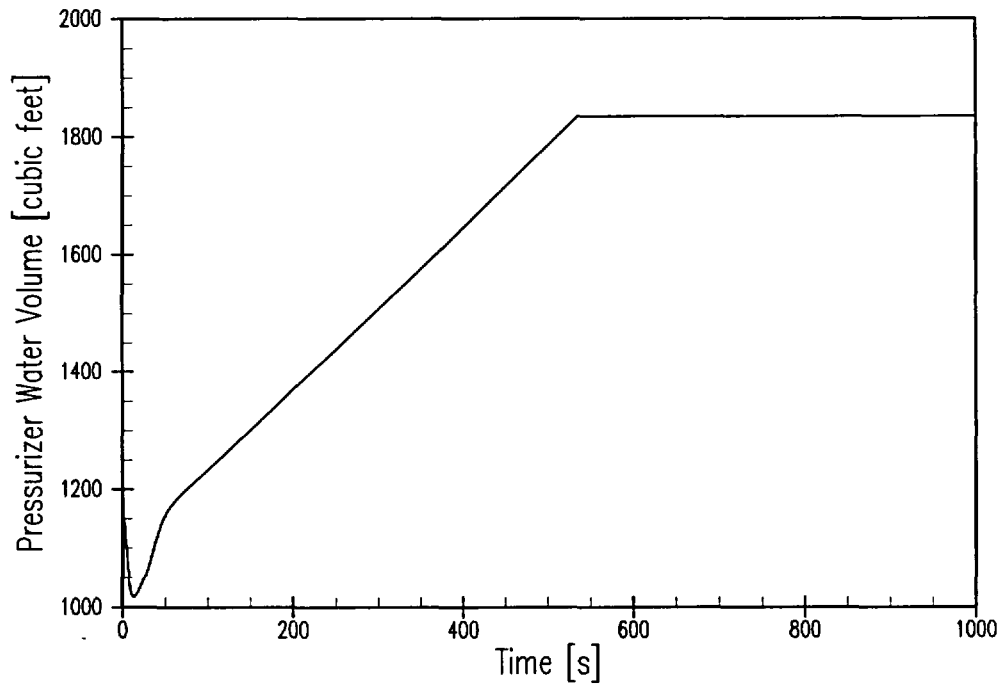
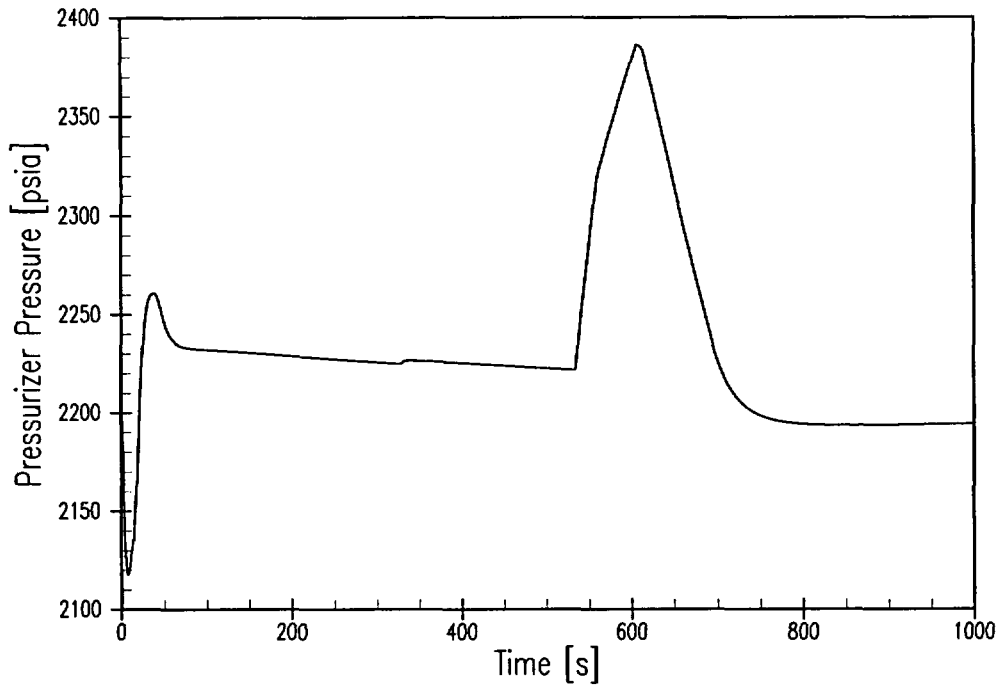


Figure 6.3.6.1-2
Inadvertent Emergency Core Cooling Systems Operation
During Power Operation
Pressurizer Pressure and Pressurizer Water Volume versus Time

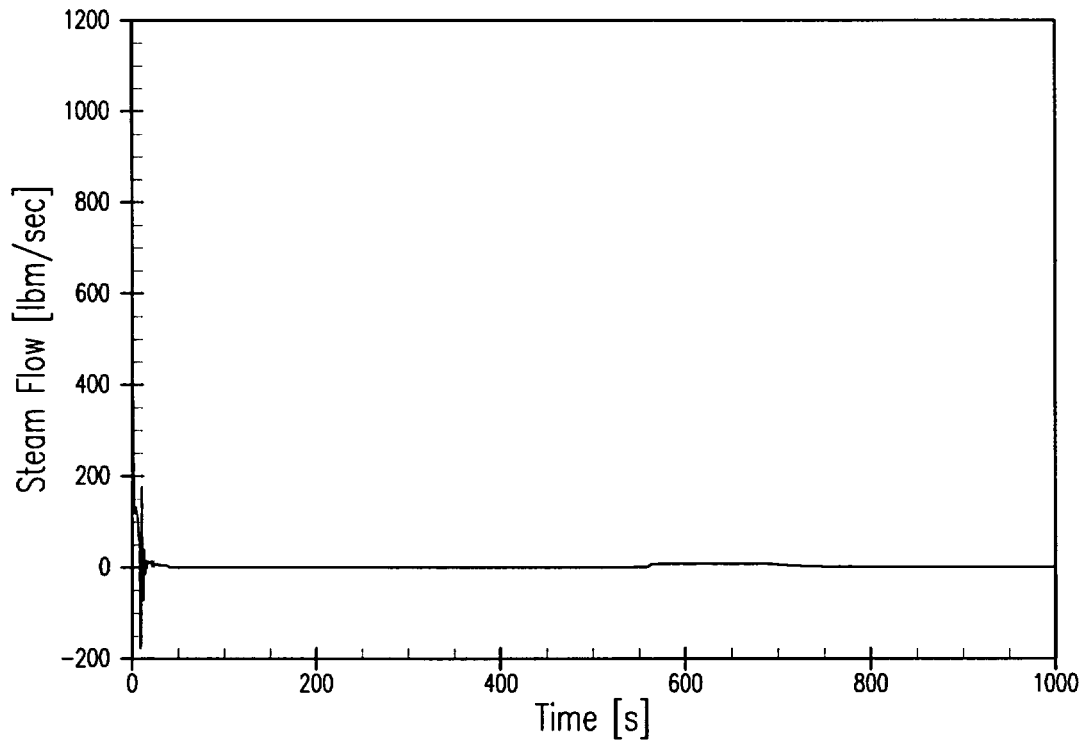


Figure 6.3.6.1-3
Inadvertent Emergency Core Cooling Systems Operation
During Power Operation
Steam Flow versus Time