February 18, 1992

Docket No. 50-443

Mr. Ted C. Feigenbaum
President and Chief Executive Officer
New Hampshire Yankee Division
Public Service Company of New Hampshire
Post Office Box 300
Seabrook, New Hampshire 03874

Dear Mr. Feigenbaum:

SUBJECT: ISSUANCE OF AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. NPF-86 - SEABROOK STATION, UNIT NO. 1 (TAC NO. M81586)

The Commission has issued the enclosed Amendment No. 9 to Facility Operating License No. NPF-86 for the Seabrook Station. This amendment is in response to your application dated September 4, 1991.

This amendment revises the Technical Specifications (TS) for the Seabrook Station by relocating several cycle-specific core operating limits from the TS to the Core Operating Limits Report (COLR). This implements the guidance of Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications."

The TS changes include: a definition for the COLR, replacement of the Radial Peaking Factor Limits Report by the COLR in Section 6.8.1.6, and revision of several individual TS sections to note that the cycle-specific parameter limits are specified in the COLR.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly FEDERAL REGISTER notice.

Sincerely,

Original signed by Gordon Edison

Gordon Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 9 to License No. NPF-86
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDI-3	:PM:PDI-3	:OGC	:D:PDI-3	:
NAME	:MRushbrook	:GEdison:mw	: MYoung	:~~WButler~~ AMENDIOLA	:
DATE	:1/21/91	:1/3/~~91~~92	:1/7/92	:2/8/92	:

OFFICIAL RECORD COPY
Document Name: SEABROOK AMEND.PKG.TAC M81586

9202270222 920218
PDR ADOCK 05000443
P PDR

DF01
1/1

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

February 18, 1992

Docket No. 50-443

Mr. Ted C. Feigenbaum
President and Chief Executive Officer
New Hampshire Yankee Division
Public Service Company of New Hampshire
Post Office Box 300
Seabrook, New Hampshire 03874

Dear Mr. Feigenbaum:

SUBJECT: ISSUANCE OF AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. NPF-86 - SEABROOK STATION, UNIT NO. 1 (TAC NO. M81586)

The Commission has issued the enclosed Amendment No. 9 to Facility Operating License No. NPF-86 for the Seabrook Station. This amendment is in response to your application dated September 4, 1991.

This amendment revises the Technical Specifications (TS) for the Seabrook Station by relocating several cycle-specific core operating limits from the TS to the Core Operating Limits Report (COLR). This implements the guidance of Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications."

The TS changes include: a definition for the COLR, replacement of the Radial Peaking Factor Limits Report by the COLR in Section 6.8.1.6, and revision of several individual TS sections to note that the cycle-specific parameter limits are specified in the COLR.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly FEDERAL REGISTER notice.

Sincerely,

G E Edison

Gordon Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 9 to
   License No. NPF-86
2. Safety Evaluation

cc w/enclosures:
See next page

cc:
Thomas Dignan, Esq.
John A. Ritsher, Esq.
Ropes and Gray
One International Place
Boston, Massachusetts 02110-2624

Dr. Mauray Tye, President
Sun Valley Association
209 Summer Street
Haverhill, Massachusetts 01831

Barbara J. Saint Andres, Esquire
Kopelman & Paige, P.C.
Counsel for Amesbury, Newburyport & Salisbury
101 Arch Street
Boston, Massachusetts 02110

Robert Backus, Esq.
Backus, Meyer and Solomon
116 Lowell Street
Manchester, New Hampshire 03106

Diane Curran, Esq.
Harmon and Weiss
2001 S Street, N.W.
Suite 430
Washington, DC 20009

Mr. Peter Brann
Assistant Attorney General
State House, Station #6
Augusta, Maine 04333

Seacoast Anti-Pollution League
5 Market Street
Portsmouth, New Hampshire 03801

Resident Inspector
U.S. Nuclear Regulatory Commission
Seabrook Nuclear Power Station
Post Office Box 1149
Seabrook, New Hampshire 03874

Mr. T. L. Harpster
Public Service Company of New Hampshire
Post Office Box 300
Seabrook, New Hampshire 03874

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Ashod N. Amirian, Esq.
145 South Main Street,
P.O. Box 38
Bradford, Massachusetts 01830

Mr. Alfred Sargent
Chairman
Board of Selectmen
Salisbury, Massachusetts 01950

Office of the Attorney General
One Ashburton Place
20th Floor
Boston, Massachusetts 02108

Durham Board of Selectmen
Town of Durham
Durham, New Hampshire 03824

Chairman, Board of Selectmen
RFD 2
South Hampton, New Hampshire 03827

R. Scott Hill-Whilton, Esquire
Lagoulis, Clark, Hill-Whilton & Rotondi
79 State Street
Newburyport, Massachusetts 01950

Board of Selectmen
Town of Amesbury
Town Hall
Amesbury, Massachusetts 01913

Paul McEachern, Esquire
25 Maplewood Avenue
P.O. Box 360
Portsmouth, New Hampshire 03801

cc:
Town of Exeter
10 Front Street
Exeter, New Hampshire 03823

Gerald Garfield, Esq.
Day, Berry and Howard
City Place
Hartford, Connecticut 06103-3499

Mr. J. F. Opeka
Northeast Utilities
P.O. Box 270
Hartford, Connecticut 06141-0270

Mr. R. M. Kacich
Northeast Utilities Service Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Jane Spector
Federal Energy Regulatory Commission
825 North Capital Street, N.E.
Room 8105
Washington, D.C. 20426

Mr. Rob Sweeney
Three Metro Center
Suite 610
Bethesda, Maryland 20814

Mr. George L. Iverson, Director
New Hampshire Office of Emergency
Management
State Office Park South
107 Pleasant Street
Concord, New Hampshire 03301

Adjudicatory File (2)
Atomic Safety and Licensing Board
Panel Docket
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Leon Maglathlin
Public Service Company of New Hampshire
1000 Elm Street
Manchester, New Hampshire 03105

Mr. Jack Dolan
Federal Emergency Management Agency
Region I
J.W. McCormack Post Office &
Courthouse Building, Room 442
Boston, Massachusetts 02109

Mr. J. M. Peschel
Public Service Company of New Hampshire
P.O. Box 300
Seabrook, New Hampshire 03874

Stephen G. Burns, Director
Office of Commission Appellate
Adjudication
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. David Rodham, Director
Massachusetts Civil Defense Agency
400 Worcester Road
P.O.Box 1496
Framingham, Massachusetts 01701-0317
ATTN: James Muckerheide

Atomic Safety and Licensing Board
Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

John P. Arnold, Attorney General
G. Dana Bisbee, Associate Attorney
General
Attorney General's Office
25 Capitol Street
Concord, New Hampshire 03301

Congressman Nicholas Mavroules
U. S. House of Representatives
70 Washington Street
Salem, Massachusetts 01970

AMENDMENT NO. 9 TO NPF-86 SEABROOK STATION DATED February 18, 1992

DISTRIBUTION:
Docket File 50-443
NRC PDR
Local PDR
PDI-3 Reading
S. Varga
J. Calvo
M. Rushbrook
W. Butler
G. Edison
OGC - 15 B18
Dennis Hagan - MNBB 3206
E. Jordan - MNBB 3701
B. Grimes - 9 A3
G. Hill (4) P1-37
Wanda Jones - 7103 MNBB
C. Grimes - 11 F23
ACRS (10) P-315
GPA/PA - 2 G5
OC/LFMB - MNBB
J. Linville, Region I
R. Lobel - 17G5
T. L. Huang - 8E23

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

PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE, ET AL.*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
License No. NPF-86

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Public Service Company of New Hampshire (the licensee), acting for itself and as agent and representative of the 11 other utilities listed below and hereafter referred to as licensees, dated September 4, 1991, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

*Public Service Company of New Hampshire is authorized to act as agent for the: Canal Electric Company, The Connecticut Light and Power Company, EUA Power Corporation, Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, New Hampshire Electric Cooperative, Inc., Taunton Municipal Light Plant, The United Illuminating Company, and Vermont Electric Generation and Transmission Cooperative, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-86 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 9 , and the Environmental Protection Plan contained in Appendix B are incorporated into Facility License No. NPF-86. PSNH shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of receipt of this letter.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 18, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 9

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Appropriate overlap pages have also been provided.

Remove	Insert
i	i
iii	iii
xiv	xiv
1-2	1-2
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
3/4 1-1	3/4 1-1
3/4 1-3	3/4 1-3
3/4 1-4	3/4 1-4
3/4 1-5	3/4 1-5
3/4 1-8	3/4 1-8
3/4 1-10	3/4 1-10
3/4 1-12	3/4 1-12
3/4 1-15	3/4 1-15
3/4 1-16	3/4 1-16
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 1-23	3/4 1-23
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1	B 3/4 2-1
B 3/4 2-3	B 3/4 2-3
6-18	6-18
----	6-18A*
----	6-18B*
----	6-18C*

*Denotes new page

INDEX

1.0 DEFINITIONS

INDEX

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.0 BASES

3.0/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

5.0 DESIGN FEATURES

6.0 ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT provides core operating limits for the current operating reload cycle. The cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.8.1.6. Plant operation within these operating limits is addressed in individual specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1.11 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation and/or injecting simulated process data to verify OPERABILITY of alarm and/or trip functions. The Digital Channel Operational Test definition is only applicable to the Radiation Monitoring Equipment.

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample with half-lives greater than 10 minutes.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.14 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.16 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.17 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.18 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.19 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.20 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain in Part A the radiological effluent sampling and analysis program and radiological environmental monitoring program. Part B of the ODCM shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.21 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.22 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

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 and Federal and State Regulations, burial ground requirements, and other requirements governing the disposal of 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 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

CONTAINMENT ENCLOSURE BUILDING INTEGRITY

1.31 CONTAINMENT ENCLOSURE BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit,

b. The Containment Enclosure Filtration System is OPERABLE, and

c. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.32 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.33 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.34 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.35 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.36 A SOURCE CHECK shall be the qualitative assessement of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.37 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.38 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.39 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.40 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.41 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.42 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.43 VENTING shall be the 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 not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, k_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 350°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3. HOT STANDBY	< 0.99	0	$\geq 350°F$
4. HOT SHUTDOWN	< 0.99	0	$350°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN for four-loop operation shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than the limiting value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limiting value:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with k_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

---

*See Special Test Exceptions Specification 3.10.1.

REACTIVITY CONTROL SYSTEMS

BORATION CONTROL

SHUTDOWN MARTIN - T_{avg} GREATER THAN 200°F

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 (Continued)

e. When in MODE 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR). Additionally, the Reactor Coolant System boron concentration shall be greater than or equal to 2000 ppm boron when the reactor coolant loops are in a drained condition.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than the limit specified in the COLR or the Reactor Coolant System boron concentration less than 2000 ppm boron, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN and boron concentration are restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limit specified in the COLR and the Reactor Coolant System boron concentration shall be determined to be greater than or equal to 2000 ppm boron when the reactor coolant loops are in a drained condition:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

    1) Reactor Coolant System boron concentration,

    2) Control rod position,

    3) Reactor Coolant System average temperature,

    4) Fuel burnup based on gross thermal energy generation,

    5) Xenon concentration, and

    6) Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less positive than 0 $\Delta k/k/°F$.

APPLICABILITY: Beginning of cycle life (BOL) limit - MODES 1 and 2* only**.
End of cycle life (EOL) limit - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR, within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

   3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.8.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With k_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

MODERATOR TEMPERATURE COEFFICIENT

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

*With k_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging pumps to the RCS.

APPLICABILITY: MODES 1, 2, and 3*

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limit specified in the CORE OPERATING LIMITS REPORT (COLR) for the above MODES at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal; and

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across the pump of greater than or equal to 2480 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first, and at least once per 31 days thereafter, except when the reactor vessel head is removed.

*An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limit specified in the CORE OPERATING LIMITS REPORT (COLR) for the above MODES at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across each pump of greater than or equal to 2480 psid is developed when tested pursuant to Specification 4.0.5.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 6,500 gallons,

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 24,500 gallons,

2) A minimum boron concentration of 2000 ppm, and

3) A minimum solution temperature of 50°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water sources shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 22,000 gallons,

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 477,000 gallons,

2) A minimum boron concentration of 2000 ppm,

3) A minimum solution temperature of 50°F, and

4) A maximum solution temperature of 98°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least the limit specified in the CORE OPERATING LIMITS REPORTS (COLR) for the above MODES at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable because of being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

3.1.3.1 ACTION b.3 (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours, except during time intervals when the rod position deviation monitor is inoperable; then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn# as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn#, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn# as specified in the COLR:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With k_{eff} greater than or equal to 1.

#The fully withdrawn position is defined as the interval within 225 to the mechanical fully withdrawn position, inclusive.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the insertion limits specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours, except during time intervals when the rod insertion limit monitor is inoperable; then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With k_{eff} greater than or equal to 1.

PAGE INTENTIONALLY BLANK

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference as specified in the CORE OPERATING LIMITS REPORT (COLR):

The indicated AFD may deviate outside the required target band specified in the COLR at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits specified in the COLR and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the required target band specified in the COLR at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AFD outside of the required target band specified in the COLR and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the required target band specified in the COLR for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits specified in the COLR and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux* ** - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

---

*See Special Test Exceptions Specification 3.10.2.

**Surveillance testing of the Power Range Neutron Flux Channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR. A total of 16 hours' operation may be accumulated with the AFD outside of the required target band as specified in the COLR during testing without penalty deviation.

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1

ACTION: (Continued)

c. With the indicated AFD outside of the required target band as specified in the COLR for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One-minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half-minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full-Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full-Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the predicted value at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

THIS PAGE INTENTIONALLY BLANK

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{.5} K(Z) \text{ for } P \leq 0.5$$

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR), and

K(Z) = the normalized $F_Q(Z)$ as a function of core height as specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit, and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased, provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

THIS PAGE INTENTIONALLY BLANK

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER,

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2b., above, to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1+PF_{xy}(1-P)],$$

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the Power Factor Multiplier for F_{xy} specified in the COLR and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 Effective Full-Power Days (EFPD), whichever occurs first.

SURVEILLANCE REQUIREMENTS

4.2.2.2d. (Continued)

2) When the F_{xy}^C is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

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

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2%, and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^C exceeding F_{xy}^L, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, 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% to account for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^{N}_{\Delta H}$ shall be less than $F^{RTP}_{\Delta H}$ [1.0 + $PF_{\Delta H}$ (1-P)].

Where: $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$F^{RTP}_{\Delta H}$ = the $F^{N}_{\Delta H}$ limit at RATED THERMAL POWER (RTP), specified in the CORE OPERATING LIMITS REPORT (COLR), and

$PF_{\Delta H}$ = the Power Factor Multiplier for $F^{N}_{\Delta H}$ specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F^{N}_{\Delta H}$ exceeding its limit:

a. Within 2 hours reduce the THERMAL POWER to the level where the LIMITING CONDITION FOR OPERATION is satisfied.

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the limit required by ACTION a., above; THERMAL POWER may then be increased, provided $F^{N}_{\Delta H}$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^{N}_{\Delta H}$ shall be demonstrated to be within its limit prior to operation above 75% RATED THERMAL POWER after each fuel loading and at least once per 31 EFPD thereafter by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% RATED THERMAL POWER.

b. Using the measured value of $F^{N}_{\Delta H}$ which does not include an allowance for measurement uncertainty.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no-load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200° F, the reactivity transients resulting from a postulated steam line break cooldown are minimal. A SHUTDOWN MARGIN as specified in the COLR and a boron concentration of greater than 2000 ppm are required to permit sufficient time for the operator to terminate an inadvertent boron dilution event with T_{avg} less than 200° F.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting end of cycle life (EOL) MTC value as specified in the COLR. The 300 ppm surveillance limit MTC value as specified in the COLR represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value as specified in the COLR.

BORATION CONTROL

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS in MODES 1, 2, or 3, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT from expected operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 22,000 gallons of 7000 ppm borated water from the boric acid storage tanks or a minimum contained volume of 477,000 gallons of 2000 ppm borated water from the refueling water storage tank (RWST).

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 provides assurance that a mass addition pressure transient can be relieved by operation of a single PORV or an RHR suction relief valve.

As a result of this, only one boron injection system is available. This is acceptable on the basis of the stable reactivity condition of the reactor, the emergency power supply requirement for the OPERABLE charging pump and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

3/4.1.2 BORATION SYSTEMS (Continued)

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT after xenon decay and cooldown from 200° F to 140° F. This condition requires a minimum contained volume of 6500 gallons of 7000 ppm borated water from the boric acid storage tanks or a minimum contained volume of 24,500 gallons of 2000 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limitations on OPERABILITY of isolation provisions for the Boron Thermal Regeneration System and the Reactor Water Makeup System in Modes 3, 4, 5, and 6 ensure that the boron dilution flow rates cannot exceed the value assumed in the transient analysis.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with rods at their individual mechanical fully withdrawn position, T_{avg} greater than or equal to 551°F and all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

The fully withdrawn position of shutdown and control banks can be varied between 225 and the mechanical fully withdrawn position (up to 232 steps), inclusive. An engineering evaluation was performed to allow operation to the 232 step maximum. The 225 to 232 step interval allows axial repositioning to minimize RCCA wear.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to

3/4.2.1 AXIAL FLUX DIFFERENCE (Continued)

RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within established limits while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours' actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

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

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA, the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

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. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

Fuel rod bowing reduces the value of DNBR. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following:

a. Design limit DNBR of 1.30 vs. 1.28,

b. Grid spacing (K_s) of 0.046 vs. 0.059,

c. Thermal diffusion coefficient of 0.038 vs. 0.059,

d. DNBR multiplier of 0.86 vs. 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}) as provided in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

When RCS $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the established limit of a measurement error of 4% for $F^N_{\Delta H}$ 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 flux maps. During normal operation the QUADRANT POWER TILT RATIO is set equal to zero once acceptability of core peaking factors has been established by review of incore maps. The limit of 1.02 is established as an indication that the power distribution has changed enough to warrant further investigation.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. Operating procedures include allowances for measurement and indication uncertainty so that the limits of 594.3°F for T_{avg} and 2205 psig for pressurizer are not exceeded.

The measurement error of 2.1% for RCS total flow rate is based upon performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is applied. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the specified limit.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.8.1.4 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the station during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the station as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity) and SOLIDIFICATION agent or absorbent (e.g., cement).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year**. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.*** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time, and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

**The dose calculations may be reported in a supplement submitted 30 days later.

***In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.8.1.4 (Continued)

to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and the ODCM, pursuant to Specifications 6.12 and 6.13, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.14. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.9 or 3.3.3.10, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.8.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

1. SHUTDOWN MARGIN limit for MODES 1, 2, 3, and 4 for Specification 3.1.1.1,

2. SHUTDOWN MARGIN limit for MODE 5 for Specification 3.1.1.2,

3. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3.1.1.3,

6.8.1.6.a (Continued)

4. Shutdown Rod Insertion limit for Specification 3.1.3.5,

5. Control Rod Insertion limits for Specification 3.1.3.6,

6. AXIAL FLUX DIFFERENCE limits and target band for Specification 3.2.1,

7. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), F_{xy}^{RTP}, and the Power Factor Multiplier for F_{xy} for Specification 3.2.2,

8. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$ and the Power Factor Multiplier for $F_{\Delta H}^{N}$ for Specification 3.2.3.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

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-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology" July 1985 (W Proprietary)

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN limit for MODES 1, 2, 3 and 4
   3.1.1.2 - SHUTDOWN MARGIN limit 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
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

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

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN limit for MODES 1, 2, 3 and 4
   3.1.1.2 - SHUTDOWN MARGIN limit for MODE 5
   3.1.1.3 - Moderator Temperature Coefficient

3. WCAP-8385-P-A, "Power Distribution Control and Load Following Procedures Topical Report," September 1974 (W Proprietary)

   Methodology for Specifications:
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE

4. WCAP-7811, "Power Distribution Control of Westinghouse Pressurized Water Reactors," December 1971 (W Proprietary)

6.8.1.6.b. (Continued)

Methodology for Specifications:
3.1.3.5 - Shutdown Rod Insertion Limit
3.1.3.6 - Control Rod Insertion Limits

5. Letter, T.M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC), January 31, 1980, Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package

   Methodology for Specification:
   3.2.1 - AXIAL FLUX DIFFERENCE

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

   Methodology for Specification:
   3.2.1 - AXIAL FLUX DIFFERENCE

7. WCAP-7308-L, "Evaluation of Nuclear Hot Channel Factor Uncertainties" December 1971 (W Proprietary)

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

8. WCAP-8622, "Westinghouse ECCS Evaluation Model, October, 1975 Version," November 1975 (W Proprietary)

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

9. WCAP-9220, "Westinghouse ECCS Evaluation Model, February 1978 Version," February 1978 (W Proprietary)

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

10. WCAP-7912-P-A, "Power Peaking Factors," January 1975 (W Proprietary)

    Methodology for Specification:
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

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

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

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

12. 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

6.8.1.6.c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. NPF-86

PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated September 4, 1991, New Hampshire Yankee (NHY) (the licensee) proposed changes to the Technical Specifications (TS) for the Seabrook Nuclear Station. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below:

(1) The Definition section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specifications 3.1.1.1., 3.1.2.2, 3.1.2.4 and 3.1.2.6

The shutdown margin limit for Modes 1, 2, 3 and 4 for this specification is specified in the COLR.

(b) Specification 3.1.1.2 and Surveillance Requirement 4.1.1.2

The shutdown margin limit for Mode 5 for this specification and for this surveillance requirement is specified in the COLR.

(c) Specification 3.1.1.3 and Surveillance Requirement 4.1.1.3

The moderator temperature coefficient (MTC) limits for this specification and for this surveillance requirement are specified in the COLR.

The technical specification 3.1.1.3 should state that the maximum upper limit shall not be more positive than 0 $\Delta k/k/^{\circ}F$.

(d) Specification 3.1.3.5 and Surveillance Requirement 4.1.3.5

The shutdown rod insertion limit for this specification and for this surveillance requirement is specified in the COLR.

(e) Specification 3.1.3.6

The control rod insertion limits for this specification are specified in the COLR.

(f) Specification 3.2.1

The axial flux difference limits and target band for this specification are specified in the COLR.

(g) Specification 3.2.2 and Surveillance Requirement 4.2.2.2

The total peaking factor (F_Q) limit at rated thermal power, the normalized F_Q limit as a function of core height and the power factor multiplier PF_{xy} for this specification and for this surveillance requirement are specified in the COLR.

(h) Specification 3.2.3

The nuclear enthalpy rise hot channel factor ($F^N \Delta H$) limit at rated thermal power and the power factor multiplier $PF_{\Delta H}$ for this specification are specified in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(3) Specification 6.8.1.6 is revised to delete a previous reporting requirement on Peaking Factor Limit Report and to add the Core Operating Limits Report to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology" July 1985 (W Proprietary)

Methodology for Specifications:

3.1.1.1 - SHUTDOWN MARGIN limit for MODES 1, 2, 3 and 4

3.1.1.2 - SHUTDOWN MARGIN limit for MODE 5

3.1.1.3 - Moderator Temperature Coefficient

3.1.3.5 - Shutdown Rod Bank Insertion Limit

3.1.3.6 - Control Rod Bank Insertion Limits

3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

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

Methodology for Specifications:

3.1.1.1 - SHUTDOWN MARGIN limit for MODES 1, 2, 3 and 4

3.1.1.2 - SHUTDOWN MARGIN limit for MODE 5

3.1.1.3 - Moderator Temperature Coefficient

(c) WCAP-8385-P-A, "Power Distribution Control and Load Following Procedures Topical Report," September 1974 (W Proprietary)

Methodology for Specifications:

3.1.3.5 - Shutdown Rod Bank Insertion Limit

3.1.3.6 - Control Rod Bank Insertion Limits

3.2.1 - AXIAL FLUX DIFFERENCE

(d) WCAP-7811, "Power Distribution Control of Westinghouse Pressurized Water Reactors," December 1971 (W Proprietary)

Methodology for Specifications:

3.1.3.5 - Shutdown Rod Bank Insertion Limit

3.1.3.6 - Control Rod Bank Insertion Limits

(e) Letter, T.M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC), January 31, 1980, Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package

Methodology for Specification:

3.2.1 - AXIAL FLUX DIFFERENCE

(f) NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981, Branch Technical Position CPB 4.3.-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

Methodology for Specification:

3.2.1 - AXIAL FLUX DIFFERENCE

(g) WCAP-7308-L, "Evaluation of Nuclear Hot Channel Factor Uncertainties," December 1971 (W Proprietary)

Methodology for Specification:

3.2.2 - Heat Flux Hot Channel Factor

(h) WCAP-8622, "Westinghouse ECCS Evaluation Model, October 1975 Version," November 1975 (W Proprietary)

Methodology for Specification:

3.2.2 - Heat Flux Hot Channel Factor

(i) WCAP-9220, "Westinghouse ECCS Evaluation Model, February 1978 Version," February 1978 (W Proprietary)

Methodology for Specification:

3.2.2 - Heat Flux Hot Channel Factor

(j) WCAP-7912-P-A, "Power Peaking Factors," January 1975 (W Proprietary)

Methodology for Specification:

3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

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

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

Methodology for Specifications:

3.1.1.1 - SHUTDOWN MARGIN limit for MODES 1, 2, 3 and 4

3.1.1.2 - SHUTDOWN MARGIN limit for MODE 5

3.1.1.3 - Moderator Temperature Coefficient

3.1.3.5 - Shutdown Rod Bank Insertion Limit

3.1.3.6 - Control Rod Bank 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

(1) 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

Items (d), (g), (h) and (i) were approved by the staff.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

A change was also made to the Action Statement of Specification 3.1.3.1. Actions b2 and c1 were changed to reference Specification 3.1.3.6 instead of Figure 3.1.1. This change was necessary because the figure has been relocated to the COLR. Consequently, this change is administrative and, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Hampshire and Massachusetts State officials were notified of the proposed issuance of the amendment. The State officials had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 57701). The amendment also involves changes to recordkeeping or reporting requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: T. L. Huang

Date: February 18, 1992