Docket file

50-443

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 23, 1994

Mr. Ted C. Feigenbaum
Senior Vice President
and Chief Nuclear Officer
North Atlantic Energy Service Corporation
Post Office Box 300
Seabrook, NH 03874

SUBJECT: AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NPF-86: WIDE-BAND OPERATION AND CORE ENHANCEMENTS - LICENSE AMENDMENT REQUEST 93-18 (TAC M87849)

Dear Mr. Feigenbaum:

The Commission has issued the enclosed Amendment No. 33 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1, (Seabrook) in response to your application dated November 23, 1993, as supplemented by letter dated August 15, 1994.

The amendment modifies the Seabrook Technical Specifications (TS) to permit operation of the Seabrook core with an expanded axial flux difference band (wide-band operation) from that currently permitted. Other TS changes allow for fuel design enhancements. Wide-band operation is based on information derived from the fixed in-core detector system (FIDS). The use of FIDS to satisfy TS requirements was approved by Amendment 27 issued on December 22, 1993. The core design enhancements are based on methodologies described in YAEC-1849P, YAEC-1854P, and YAEC-1856P which were approved previously for use at Seabrook. North Atlantic supported the proposed technical specification changes with reanalyses of the UFSAR Chapter 15 accidents and transients, documented in YAEC-1871, and a Westinghouse Electric Corporation loss-of-coolant-accident reanalysis. These supporting documents and a revised Core Operating Limits Report were submitted with the application for amendment.

North Atlantic proposed certain changes to TS 3.1.1.3 that mould have permitted operation with a positive moderator temperature coefficient. The Commission has not yet determined if this change is acceptable. We have discussed with your staff additional information that North Atlantic must provide for us to continue with our review of this issue. Therefore, the proposed change to TS 3.1.1.3 is not being approved at this time. Additionally, North Atlantic will need to modify TS 6.8.1.6, regarding the COLR prior to entering Cycle 5 to reflect the results of these evaluations and the new methodologies used to develop the COLR.

DF01 CP-1

The amendment affects TS Sections 3.1.3.4, 3.2.1, 3.2.2, 3.2.3, 3.2.4, 3.2.5, 3.3.3.2, 4.2.1, 4.2.2, 4.2.5, 4.5.2, 5.3, and 6.8.1, Figure 2.1-1, and Tables 2.2-1, 3.3-4, and 4.3-1.

Our Safety Evaluation relating to the proposed TS changes and supporting analyses is enclosed. The Notice of Issuance, which has been forwarded to the Office of the Federal Register for publication, is also enclosed.

Sincerely,

Original signed by:

Albert W. De Agazio, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-443
Serial No. SEA-94-025

Enclosures: 1. Amendment No. 33 to NPF-86
2. Safety Evaluation related to Amendment 33
3. Notice

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTH ATLANTIC ENERGY SERVICE CORPORATION, ET AL*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. NPF-86

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

   A. The application for amendment by North Atlantic Energy Service Corporation, et al. (the licensee), dated November 23, 1993, as supplemented by letter dated August 15, 1994, complies 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;

   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.

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*North Atlantic Energy Service Company (NAESCO) is authorized to act as agent for the: North Atlantic Energy Corporation, Canal Electric Company, The Connecticut Light and Power Company, Great Bay Power Corporation, Hudson Light and Power Department, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, New Hampshire Electric Cooperative, Inc., Taunton Municipal Light Plant, and The United Illuminating Company, 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. 33, and the Environmental Protection Plan contained in Appendix B are incorporated into Facility License No. NPF-86. NAESCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented before startup from the fourth refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip F. McKee

Phillip F. McKee, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 23, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 33

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

Replace the following pages of Appendix A, Technical Specifications, with the attached pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Overleaf pages have been provided.*

Remove	Insert
1-1*	1-1*
1-2	1-2
2-1*	2-1*
2-2	2-2
2-3*	2-3*
2-4	2-4
2-5	2-5
2-6*	2-6*
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
B 2-1	B 2-1
B 2-2	B 2-2
B 2-5	B 2-5
B 2-6*	B 2-6*
3/4 1-19*	3/4 1-19*
3/4 1-20	3/4 1-20
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
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 3-11*	3/4 3-11*
3/4 3-12	3/4 3-12
3/4 3-23*	3/4 3-23*
3/4 3-24	3/4 3-24
3/4 3-27*	3/4 3-27*
3/4 3-28	3/4 3-28
3/4 3-39*	3/4 3-39*
3/4 3-40	3/4 3-40
3/4 5-5*	3/4 5-5*
3/4 5-6	3/4 5-6

3/4 5-7	3/4 5-7
3/4 5-8*	3/4 5-8*
B 3/4 1-1*	B 3/4 1-1*
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 3-3*	B 3/4 3-3*
B 3/4 3-4	B3/4 3-4
5-9	5-9
5-10*	5-10*
6-17*	6-17*
6-18	6-18
6-18A	6-18A
6-18B	6-18B
6-18C	6-18C

41.0 DEFINITIONS

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 (COLR) 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.

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for four-loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.6.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.6.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.6.

NOTE: FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

NO TEXT WILL PRINT

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Value column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R + S \leq TA$

   Where:

   Z = The value from Column Z of Table 2.2-1 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2-1
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	1.42	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	1.42	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤31.1% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.6 \times 10^5$ cps
7. Overtemperature ΔT	N.A	N.A	N.A	See Note 1	See Note 2
8. Overpower ΔT	N.A	N.A	N.A	See Note 3	See Note 4
9. Pressurizer Pressure - Low	N.A	N.A	N.A	≥1945 psig	≥1,933 psig
10. Pressurizer Pressure - High	N.A	N.A	N.A	≤2385 psig	≤2,397 psig

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. Pressurizer Water Level - High	8.0	4.20	0.84	≤92% of instrument span	≤93.75% of instrument span
12. Reactor Coolant Flow - Low	2.5	1.9	0.6	≥90% of measured loop flow	≥89.3% of measured loop flow
13. Steam Generator Water Level Low - Low	14.0	12.53	0.55	≥14.0% of narrow range instrument span	≥12.6% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	15.0	1.39	0	≥10,200 volts	≥9,822 volts
15. Underfrequency - Reactor Coolant Pumps	2.9	0	0	≥55.5 Hz	≥55.3 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥500 psig	≥450 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 2.2-1 (continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤10% of RTP*	≤12.1% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	≤10% RTP* Turbine Impulse Pressure Equivalent	≤12.3% of RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤50% of RTP*	≤52.1% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤20% of RTP*	≤22.1% of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	≥10% of RTP*	≥7.9% of RTP*
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	≤10% RTP* Turbine Impulse Pressure Equivalent	≤12.3% RTP* Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by RTD Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	Value specified in the COLR;
K_2	=	Value specified in the COLR;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in lead-lag compensator for T_{avg}, $\tau_4 \geq 33$ s, $\tau_5 \leq 4$ s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T^1 ≤ 588.5°F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = Value specified in COLR;

P = Pressurizer pressure, psig;

P^1 = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

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

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

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where:

ΔT	=	As defined in Note 1,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1,
τ_1, τ_2	=	As defined in Note 1,
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	=	Value specified in the COLR,
K_5	=	Value specified in the COLR,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_7	=	Time constants utilized in rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = Value specified in COLR,

T = As defined in Note 1,

T^{11} = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F),

S = As defined in Note 1, and

$f_2(\Delta I)$ = A function of the indicated difference between the top and bottom detectors of the power-range neutron ion chambers as specified in the COLR.

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

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and, therefore, THERMAL POWER and reactor coolant temperature and pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNB design basis is as follows: uncertainties in the DNBR correlation, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with 95 percent confidence level that DNB will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit value, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy rise hot channel factor $F^N_{\Delta H}$, at RATED THERMAL POWER, of 1.65. The value of $F^N_{\Delta H}$ at reduced power is assumed to vary according to the expression:

$$F^N_{\Delta H} = 1.65 \, [1+ 0.3 \, (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

This expression conservatively bounds the cycle specific limits on $F^N_{\Delta H}$ specified in Technical Specification 3/4.2.3 and the COLR. The Safety Limits in Figure 2.1-1 are also based on a reference cosine axial power shape with a peak of 1.55.

2.1.1 REACTOR CORE (Continued)

The resulting heat flux conditions are more limiting than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion, assuming the axial power imbalance is within the limits of the $f_1(\Delta I)$ and $f_2(\Delta I)$ functions of the Overtemperature and Overpower ΔT trips. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT and Overpower ΔT trips will reduce the setpoints to provide protection consistent with core safety limits for cycle specific power distribution.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants, which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is, therefore, consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3110 psig) of design pressure to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), pressure is within the range between the Pressurizer High and Low Pressure trips and power is less than the Overpower ΔT trip setpoint. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, and (3) axial power distribution to ensure that the allowable heat generation rate (Kw/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip, thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure that could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full-power equivalent); and on increasing power, the Pressurizer High Water Level trip is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 50% of RATED THERMAL POWER), an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7, an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Emergency Feedwater System.

MOVABLE CONTROL ASSEMBLIES

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full range of rod travel at least once per 18 months.

*With the Reactor Trip System breakers in the closed position.
**See Special Test Exceptions Specification 3.10.5

MOVABLE CONTROL ASSEMBLIES

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the mechanical fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} for each loop greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System that could affect the drop time of those specific rods, and

c. At least once per 18 months.

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:

a. The limits specified in the COLR, with the Fixed Incore Detector (FIDS) Alarm OPERABLE, or

b. The limits specified in the COLR, when the FIDS Alarm is inoperable.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER.

ACTION:

a. With the indicated AFD* outside of the applicable limits specified in the COLR:

1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours, and

3. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

b. With an OPERABLE FIDS Alarm exceeding in limit:

1. Comply with the AFD limits specified in the COLR for operation with the FIDS Alarm inoperable within 15 minutes and,

2. Verify THERMAL POWER is less than the maximum power limit established by Surveillance Requirement 4.2.1.2 within 15 minutes and,

3. Identify and correct the cause of the FIDS Alarm prior to operation beyond the limits specified in the COLR for operation with the FIDS Alarm inoperable.

c. With the FIDS Alarm inoperable, within 4 hours,

1. Comply with the AFD limits specified in the COLR for operation with the FIDS Alarm inoperable, and

2. Verify THERMAL POWER is less than the maximum power limit established by Surveillance Requirement 4.2.1.2.

*The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% 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 At least once per 31 EFPD determine the maximum allowed power for operation with the FIDS Alarm inoperable by comparing $F_Q(Z)$ to the $F_Q(Z)$ limit established for operation with the FIDS Alarm inoperable.

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

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

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

APPLICABILITY: MODE 1.

ACTION:

a. With $F_Q(Z)$ exceeding its limit:

1. 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

2. 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.

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

LIMITING CONDITION FOR OPERATION

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be demonstrated to be within its limits prior to operation above 75% RATED THERMAL POWER after each fuel loading and at least once per 31 EFPD thereafter by:

a. Using the Incore Detector System to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% when using the movable incore detectors or 5.21% when using the fixed incore detectors, to account for measurement uncertainties.

4.2.2.3 The limits of Specification 3.2.2 are not applicable in the following core plane 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.

4.2.2.4 Each fixed incore detector alarm setpoint shall be updated at least once per 31 EFPD. The alarm setpoints will be based on the latest available power distribution, so that the alarm setpoint does not exceed the $F_Q(Z)$ limit defined in Technical Specification 3.2.2.

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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 the limits specified in the COLA.

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 Incore Detector System 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.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the Incore Detector System to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by either:

a. Using the four pairs of symmetric detector locations or

b. Using the Incore Detector System to monitor the QUADRANT POWER TILT RATIO subject to the requirements of Specification 3.3.3.2.

*See Special Test Exceptions Specification 3.10.2

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}, ≤ 594.3°F

b. Pressurizer Pressure, ≥ 2185 psig*

c. Reactor Coolant System Flow shall be:

1. ≥ 382,800 gpm**; and,

2. ≥ 392,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
Reactor Trip System Interlocks (Continued)						
e. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(7, 14), R(15)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be closed and the Control Rod Drive System is capable of rod withdrawal.

**Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

***Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 50% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purposes of this surveillance requirement, monthly shall mean at least once per 31 EFPD.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated and compared to the initial curves. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purposes of this surveillance requirement, quarterly shall mean at least once per 92 EFPD.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) (Not used)

(9) Surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

ACTION STATEMENTS (Continued)

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generator, Phase "A" Isolation, Containment Ventilation Isolation, and Emergency Feedwater, Service Water to Secondary Component Cooling Water Isolation, CBA Emergency Fan/Filter Actuation, and Latching Relay).					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--Hi-1	4.2	0.71	1.67	≤ 4.3 psig	≤ 5.3 psig
d. Pressurizer Pressure--Low	N.A.	N.A.	N.A.	≥ 1800 psig	≥ 1786 psig
e. Steam Line Pressure--Low	13.1	10.71	1.63	≥ 585 psig	≥ 568 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--Hi-3	3.0	0.71	1.67	≤ 18.0 psig	≤ 18.7 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Emergency Feedwater					
a. Manual Initiation					
(1) Motor driven pump	N.A.	N.A.	N.A.	N.A.	N.A.
(2) Turbine driven pump	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low Start Motor-Driven Pump and Start Turbine-Driven Pump	14.0	12.53	0.55	≥ 14.0% of narrow range instrument span.	≥ 12.6% of narrow range instrument span.
d. Safety Injection Start Motor-Driven Pump and Turbine-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9. for Loss-of-Offsite Power Setpoints and Allowable Values.				
8. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low	2.75	1.0	1.8	≥122,525 gals.	≥121,609 gals.
Coincident With Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Loss of Power (Start Emergency Feedwater)					
a. 4.16 kV Bus E5 and E6 Loss of Voltage	N.A.	N.A.	N.A.	≥ 2975 volts with a ≤ 1.20 second time delay.	≥ 2908 volts with a ≤ 1.315 second time delay.
b. 4.16 kV Bus E5 and E6 Degraded Voltage	N.A.	N.A.	N.A.	≥ 3933 volts with a ≤ 10 second time delay.	≥ 3902 volts with a ≤ 10.96 second time delay.
Coincident with: Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1950 psig	≤ 1962 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment - Post LOCA - Area Monitor	S	R	Q	All
b. RCS Leakage Detection				
1) Particulate Radioactivity	S	R	Q	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	Q	1, 2, 3, 4
2. Containment Ventilation Isolation				
a. On Line Purge Monitor	S	R	Q	1, 2, 3, 4
b. Manipulator Crane Area Monitor	S	R	Q	6#
3. Main Steam Line	S	R	Q	1, 2, 3, 4
4. Fuel Storage Pool Areas				
a. Radioactivity-High-Gaseous Radioactivity	S	R	Q	*
5. Control Room Isolation				
a. Air Intake Radiation Level				
1) East Air Intake	S	R	Q	All
2) West Air Intake	S	R	Q	All
6. Primary Component Cooling Water				
a. Loop A	S	R	Q	All
b. Loop B	S	R	Q	All

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.

During CORE ALTERNATIONS or movement of irradiated fuel within the containment.

INSTRUMENTATION

MONITORING INSTRUMENTATION

INCORE DETECTOR SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Incore Detector System shall be OPERABLE with:

a. At least 75% of the detector locations and,

b. A minimum of two detector locations per core quadrant,

c. An OPERABLE incore detector location consist of a fuel assembly containing a fixed detector string with a minimum of three OPERABLE detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the Incore Detector System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^{M}_{\Delta H}$ and $F_{Q}(Z)$, or

d. Input into the FIDS Alarm

ACTION:

With the Incore Detector System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

(Plant procedures are used to determine that the Incore Detector System is OPERABLE.)

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
SI-V-3	Accumulator Isolation	Open*
SI-V-17	Accumulator Isolation	Open*
SI-V-32	Accumulator Isolation	Open*
SI-V-47	Accumulator Isolation	Open*
SI-V-114	SI Pump to Cold-Leg Isolation	Open
RH-V-14	RHR Pump to Cold-Leg Isolation	Open
RH-V-26	RHR Pump to Cold-Leg Isolation	Open
RH-V-32	RHR to Hot-Leg Isolation	Closed
RH-V-70	RHR to Hot-Leg Isolation	Closed
SI-V-77	SI to Hot-Leg Isolation	Closed
SI-V-102	SI to Hot-Leg Isolation	Closed

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) 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.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing primary CONTAINMENT INTEGRITY, and

2) At least once daily of the areas affected within containment by containment entry and during the final entry when primary CONTAINMENT INTEGRITY is established.

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*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

d. At least once per 18 months by:

1) Verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 365 psig, the interlocks prevent the valves from being opened.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on (Safety Injection actuation and Automatic Switchover to Containment Sump) test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump, ≥ 2480 psid;

2) Safety Injection pump, ≥ 1445 psid; and

3) RHR pump, ≥ 171 psid.

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

High Head SI System Valve Number	Intermediate Head SI System Valve Number
SI-V-143	SI-V-80
SI-V-147	SI-V-85
SI-V-151	SI-V-104
SI-V-155	SI-V-109
	SI-V-117
	SI-V-121
	SI-V-125
	SI-V-129

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 306 gpm, and

b) The total pump flow rate is less than or equal to 549 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 419 gpm, and

b) The total pump flow rate is less than or equal to 669 gpm.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 4213 gpm.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3.1 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

BASES

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 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.

Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, the MTC is measured as required by Surveillance Requirement 4.1.1.3.a. A measurement bias is derived from the difference between test measurement and test prediction. All predicted values of MTC for the cycle are conservatively corrected based on measurement bias. The corrected predictions are then compared to the maximum upper limit of Technical Specification 3.1.1.3. Control rod withdrawal limits are established, if required, to assure all corrected values of predicted MTC will be less positive than the maximum upper limit required by Technical Specification 3.1.1.3.

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 boron 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.

3/4.2 POWER DISTRIBUTION LIMITS

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 the design DNBR value 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.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) specified in the CORE OPERATING LIMITS REPORT (COLR) assure that the design limits on peak local power density and minimum DNBR are not exceeded during normal operation and the consequences of any Non-LOCA event would be within specified acceptance criteria.

For operation with the Fixed Incore Detectors (FIDS), assurance that the $F_Q(Z)$ limit of Specification 3.2.2 is not exceeded during either normal operation or in the event of xenon redistribution following power changes is provided by a separate Fixed Incore Detector Alarm through the plant process computer. A FIDS Alarm will be generated when a predetermined number of individual detectors exceed their alarm setpoint. The setpoint for each individual detector is adjusted by the normal 5.21% for system measurement uncertainty and 3% for engineering uncertainty. This assures that the consequences of a LOCA would be within specified acceptance criteria.

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 limits specified in the COLR. These alarms are active when power is greater than 50% of RATED THERMAL POWER.

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 design limit DNBR includes margin to offset any rod bow penalty. Margin is also maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is available for plant design flexibility.

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

For operation with the Fixed Incore Detector System (FIDS) Alarm OPERABLE, the cycle-dependent normalized axial peaking factor, K(Z), specified in COLR accounts for axial power shape sensitivity in the LOCA analysis. Assurance that the $F_Q(Z)$ limit on Specification 3.2.2 is met during both normal operation and in the event of xenon redistribution following power changes is provided by the FIDS Alarm through the plant process computer. This assures that the consequences of a LOCA would be within specified acceptance criteria.

For operation with the FIDS Alarm inoperable, the cycle-dependent normalized axial peaking factor, K(Z), specified in COLR accounts for possible xenon redistribution following power changes in addition to axial power shape sensitivity in the LOCA analysis. This assures that the consequences of a LOCA would be within specified acceptance criteria.

When RCS $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the established limit. A bounding measurement error of 4.13% 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 Detector System surveillances. 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 surveillances. 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 updated FSAR assumptions and have been analytically demonstrated adequate to assure compliance with acceptance criteria for each analyzed transient. Operating procedures include allowances for measurement and indication uncertainty so that the limits of 594.3°F for T_{avg} and 2185 psig for pressurizer pressure are not exceeded.

RCS flow must be greater than or equal to, 1) the Thermal Design Flow (TDF) with an allowance for measurement uncertainty and, 2) the minimum measured flow used in place of the TDF in the analysis of DNB related events when the Revised Thermal Design Procedure (RTDP) methodology is utilized.

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.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) emergency feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) automatic service water valves position.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure, and the manual block of SI and steamline isolation on steamline low pressure. On the manual block of steamline low pressure, manual block of steamline low pressure automatically initiates steamline isolation on steam generator pressure negative rate - high.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and all feedwater isolation valves; inhibits feedwater control valve modulation; and blocks the start of the startup feedwater pump.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents

MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS (Continued)

and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 INCORE DETECTOR SYSTEM

The Incore Detector System consists of either a) fixed detector strings and their associated signal processing, or b) movable incore detectors and their associated signal processing. OPERABILITY may be met by either fixed detectors or movable detectors but not by a combination of both.

The OPERABILITY of the Incore Detector System ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore detectors may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of Appendix A to 10 CFR Part 50.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 52.0 psig and a temperature of 296°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with a zirconium alloy. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.15 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 57 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,265 cubic feet at a nominal T_{avg} of 588.5°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes margin for uncertainty in calculation methods and mechanical tolerances with a 95% probability at a 95% confidence level.

b. A nominal 10.35 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes margin for uncertainty in calculational methods and mechanical tolerances with a 95% probability at a 95% confidence level.

b. A k_{eff} equivalent to less than or equal to 0.98 when aqueous foam moderation is assumed, which includes margin for uncertainty in calculational methods and mechanical tolerances with a 95% probability at a 95% confidence level.

c. A nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 14 feet 6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1236 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.8.1.4 A routine Annual Radioactive Effluent Release Report covering the operation of the station during the previous calendar year of operation shall be submitted by May 1 of each year.

The Annual 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 Annual Radioactive Effluent Release Report 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 Annual Radioactive Effluent Release Report 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

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*In lieu of submission with the Annual 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.

ANNUAL 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 Annual Radioactive Effluent Release Report 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 Annual Radioactive Effluent Release Report 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 Annual Radioactive Effluent Release Report 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. Cycle dependent Overpower ΔT and Overtemperature ΔT trip setpoint parameters and function modifiers for operation with skewed axial power profiles for Table 2.2-1 of Specification 2.2.1,

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

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

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

6.8.1.6.a. (Continued)

5. Shutdown Rod Insertion limit for Specification 3.1.3.5,

6. Control Rod Bank Insertion limits for Specification 3.1.3.6,

7. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1,

8. Heat Flux Hot Channel Factor, F_Q^{RTP} and K(Z) for Specification 3.2.2,

9. Nuclear Enthalpy Rise Hot Channel Factor, and $F_{\Delta H}^{RTP}$ 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-10266-P-A, Rev. 2 with Addenda (Proprietary) and WCAP-11524-A (Nonproprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", August, 1986

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

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

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

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

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

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

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

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

6.8.1.6.b. (Continued)

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

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

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

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

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

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

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

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

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

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

6.8.1.6.b. (Continued)

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

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

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

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

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. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NO. NPF-86

NORTH ATLANTIC ENERGY SERVICE CORPORATION

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated November 23, 1993 (Ref. 1), North Atlantic Energy Service Corporation (North Atlantic), proposed changes to the Seabrook Station, Unit No. 1 (Seabrook) Technical Specifications (TS) that would permit core operation with an expanded axial-flux-difference (wide-band) compared to the current constant-axial-offset. Other proposed TS changes would allow for fuel design enhancements. Additional supporting and clarifying information was submitted on August 15, 1994 (Ref. 2).

Wide-band operation is made possible through the use of the fixed in-core detector system (FIDS). The number and type of detectors used in the FIDS have been described in Yankee Atomic Electric Company report, YAEC-1855P (Ref. 3). The use of the FIDS to satisfy TS requirements for neutron flux measurement was authorized by Amendment 27 to Operating License No. NPF-86. The proposed TS changes were based on core reanalyses using new methodologies described in three reports (Refs. 4 - 6) and their corresponding NRC approvals are in references 7 - 9 respectively. The proposed core design enhancements refer to fuel design changes which will improve fuel utilization.

North Atlantic's submittals supporting the proposed changes included reanalyses of those transients and accidents which are discussed in the Updated Final Safety Analysis Report (UFSAR). The choices of initial conditions and approximations for the reanalyses of the non-LOCA transients and the reanalyses results are documented in YAEC-1871, "Safety Analysis in Support of Wide-Band Operation and Core Design Enhancements for Seabrook Station" (Ref. 10). Completeness of the Chapter 15 analyses is shown in Table 1-1. The small-break Loss-of-Coolant-Accident (LOCA) and large-break LOCA reanalyses and a revised core operating limits report (COLR) also were included with Ref. 1.

COMPARISON OF ANALYZED TRANSIENTS
STANDARD REVIEW PLAN (NUREG 0800) VERSUS YAEC-1871

Standard Review Plan (NUREG 0800) Transients	YAEC-1871 Transients
15.1.1, 15.1.2, 15.1.3, 15.1.4 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Secondary Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	5.1.1 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature 5.1.2 Feedwater System Malfunctions Causing an Increase in Feedwater Flow 5.1.3 Excessive Increase in Secondary Steam Flow 5.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve
15.1.5 Steam System Piping Failures	5.1.5 Steam System Piping Failure
15.2.1 - 15.2.5 Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)	5.2.1 Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow 5.2.2 Loss of External Load 5.2.3 Turbine Trip 5.2.4 Inadvertent Closure of Main Steam Isolation Valves 5.2.5 Loss of Condenser Vacuum and Other Events Resulting in a Turbine Trip
15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries	5.2.6 Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)
15.2.7 Loss of Normal Feedwater Flow	5.2.7 Loss of Normal Feedwater Flow
15.2.8 Feedwater System Pipe Breaks	5.2.8 Feedwater System Pipe Break
15.3.1 - 15.3.2 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	5.3.1 Partial Loss of Forced Reactor Coolant Flow 5.3.2 Complete Loss of Forced Reactor Coolant Flow
15.3.3 - 15.3.4 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	5.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor) 5.3.4 Reactor Coolant Pump Shaft Seizure (Locked Rotor) Followed by Loss of Offsite Power 5.3.5 Reactor Coolant Pump Shaft Break
15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition	5.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition
15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	5.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power
15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)	5.4.3 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)

COMPARISON OF ANALYZED TRANSIENTS (CONTINUED)			
Standard Review Plan (NUREG 0800) Transients		YAEC-1871 Transients	
15.4.4 -15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	5.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature
15.4.6	Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant	5.4.5	Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	5.4.6	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
15.4.8	Spectrum of Rod Ejection Accidents	5.4.7	Spectrum of Rod Cluster Control Assembly Ejection Accidents
15.4.9	Spectrum of Rod Drop Accidents (BWR)		Not Applicable
15.5.1 - 15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	5.5.1 5.5.2	Inadvertent Operation of ECCS During Power Operation Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	5.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	5.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment
15.6.3	Radiological Consequences of Steam Generator Tube Failure	5.6.3	Steam Generator Tube Rupture
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)		Not Applicable
15.6.5	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	5.6.4	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
15.7.1	Waste Gas System Failure	5.7	Radioactive Release From a System or Component
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	5.7	Radioactive Release From a System or Component
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	5.7	Radioactive Release From a System or Component
15.7.4	Radiological Consequences of Fuel Handling Accidents	5.7	Radioactive Release From a System or Component
15.7.5	Spent Fuel Cask Drop Accidents	5.7	Radioactive Release From a System or Component
15.8	Anticipated Transients Without Scram	5.8	Anticipated Transients Without Scram

in YAEC-1871, and a Westinghouse Electric Corporation loss-of-coolant-accident reanalysis. These supporting documents and a revised Core Operating Limits Report were submitted with the application for amendment.

The licensee proposed certain changes to TS 3.1.1.3 that would permit operation with a positive moderator temperature coefficient. The Commission has not yet determined the acceptability of this proposed change pending submission of additional information from the licensee. Therefore, the proposed change to TS 3.1.1.3 is not implemented by this amendment.
This amendment affects TS Sections 3.1.3.4, 3.2.1, 3.2.2, 3.2.3, 3.2.4, 3.2.5, 3.3.3.2, 4.2.1, 4.2.2, 4.2.5, 4.5.2, 5.3, and 6.8.1, Figure 2.1-1, and Tables 2.2-1, 3.3-4, and 4.3-1.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on January 18, 1994 (59 FR 2632). No request for hearing or petition for leave to intervene was filed following this notice.

For further details with respect to this action see (1) the application for amendment dated November 23, 1993, as supplemented by letter dated August 15, 1994, (2) Amendment No. 33 to License No. NPF-86, and (3) the Commission's related Safety Evaluation, and (4) the Commissions Environmental Assessment dated September 27, 1994. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Exeter Public Library, 47 Front Street, Exeter, NH 03833.

Dated at Rockville, Maryland this 23rd day of November 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Albert W. De Agazio

Albert W. De Agazio, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

The following is a summary of the changes proposed by North Atlantic:[1]

- Increased core power-distribution peaking factors; F_Q: from 2.32 to 2.50, and $F_{\Delta H}$: from 1.55 to 1.65

- Implementation of fuel design enhancements
  - low ΔP Zircaloy grids
  - zirlo cladding

- Deletion of thimble plugs

- Allowance for up to 8 percent steam generator tube plugging

- Implementation of relaxed surveillance parameters
  - Pressurizer pressure uncertainty: from ±30 psi to ±50 psi
  - Low reactor coolant system (RCS) pressure SI setpoints: from 1760 psia to 1665 psia
  - Low RCS pressure SI time delay: from 27 to 30 seconds
  - Emergency feedwater temperature: from 88°F to 100°F
  - Emergency feedwater time delay: from 60 to 75 seconds

- Implement wide-band axial flux difference (ΔI) operation
  - When FIDS is operable:
    Constant axial offset control (CAOC) limits on axial flux difference is replaced with ΔI Limiting Condition for Operation (LCO) for departure from nucleate boiling (DNB) and fuel centerline melt per YAEC-1854P, and the linear heat generation rate (LHGR) LOCA limit
  - When FIDS is inoperable:
    CAOC limits on axial flux difference is replaced with ΔI LCO and power distribution surveillance (equivalent to Westinghouse relaxed axial offset)

- Application of the revised thermal design procedure and WRB-1 to provide additional thermal margin per YAEC-1849P (VIPRE-01)

- Relocate certain parameters from the TS to the Core Operating Limits Report (COLR)
  - Cycle dependent overpower-ΔT and overtemperature-ΔT trip setpoint parameters and function modifiers (from Table 2.1-1)
  - K(z) limits for operation with FIDS operable and FIDS inoperable and the new burnup dependent $F_{\Delta HN}$ limits.

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[1]Review of a proposed positive MTC has been deferred for a separate review and until additional information is submitted.

2.0 TRANSIENT AND NON-LOCA ACCIDENTS

2.1 Background

North Atlantic's license amendment application included Yankee Atomic Electric Company report, YAEC-1871, "Safety Analysis in Support of Wide-Band Operation and Core Design Enhancements for Seabrook Station." The analyses in YAEC-1871 were submitted to support proposed Seabrook operation with expanded axial flux difference limiting condition for operation (LCO) and with enhanced core and system design features.

YAEC-1871 presents the results of accident analyses (except both the small and large break LOCA analyses) for Seabrook for cycle 5 and subsequent cycles and provides a summary of the analytical methods and core design parameters and assumptions used in the reanalyses of the UFSAR accidents and transients for Seabrook. Also included is a discussion of the impact of the new analyses on the core thermal and hydraulic design.

The transients are grouped as follows:

- Increase in heat removal by the secondary system
- Decrease in heat removal by the secondary system
- Decrease in RCS flow rate
- Reactivity and power distribution anomalies
- Increase in reactor coolant inventory
- Decrease in reactor coolant inventory
- Radioactive release from a system or component
- Anticipated transients without scram.

2.2 ANALYSIS METHODOLOGY AND APPLICATIONS

The thermal-hydraulic subchannel analysis was performed with the VIPRE-01 methodology using the WRB-1 DNB correlation and the system thermal-hydraulic transient analysis was performed using the RETRAN code. The new setpoint methodology is based on YAEC-1854P, which includes consideration of the use of the FIDS to monitor the core power distribution continuously for compliance to the applicable LCOs. The axial flux (power) difference LCO is defined by power distributions which assure adequate margin for thermal design limits DNB and centerline melt. YAEC-1854P describes the determination of the axial flux difference LCO band and the associated system setpoints for the overtemperature-ΔT and overpower-ΔT trips. Thus, the analysis methodology is described in NRC approved topical reports (Refs. 4 - 6).

In addition to using the WRB-1 correlation, the proposed methodology and TS changes provide for (1) increased core power distribution peaking factors, (2) thimble plug deletion, (3) increased steam generator tube plugging, (4) implementation of new fuel designs (such as low pressure drop Zircaloy grids), (5) zirlo cladding, (6) modification of some analysis assumptions related to certain surveillance parameters such as low pressurizer pressure safety injection actuation setpoint and time delay, and (7) expansion of the axial flux difference LCO band.

The heat flow hot channel factor, F_Q, is increased to 2.50 from 2.32 and the enthalpy rise hot channel factor (at rated thermal power), $F_{\Delta H}{}^{N}$, is increased to 1.65 from 1.55. In the expression,

$$F_{\Delta H}{}^{N}=F_{\Delta H}{}^{RTP}(1.0+PF_{\Delta H}(1.0-P))$$

where $F_{\Delta H}{}^{RTP}$ is at rated thermal power, the factor, $PF_{\Delta H}$, is increased from 0.2 to 0.3. These values are used in the LOCA analysis and in the derivation of the cycle-independent thermal limits in TS Fig. 2.1.1. YAEC-1871 supports these values which are specified in the COLR.

With the introduction of the FIDS, the thimble plugs may be removed at some time in the future. This creates the possibility of increased bypass flow. The reanalysis provides for an increase in bypass flow to 7.5 percent from 5.8 percent.

Presently, Seabrook has a negligible number of steam generator tubes plugged, however, the reanalysis provides for up to 8 percent tube plugging. This allowance is implemented by a reduction in the steam generator heat transfer by 8 percent, a 2 percent reduction in the RCS flow for the analysis of DNB events and a 2 percent reduction in the thermal design flow. The minimum measured flow is specified in TS 3.2.5.

Several changes are implemented to accommodate future fuel design changes:

a. Control rod drop time is increased to 2.4 seconds from 2.2 seconds. The increased control rod drop time accommodates new fuel and grid designs which have a slight reduction in control rod guide tube inside diameter, which may cause rod drop time increase. The 2.4 second drop time has been used in the reanalyses. TS 3.1.3.4 is changed accordingly.

b. Pressurizer pressure uncertainty has been increased from ±30 psi to ±50 psi. This change was implemented to facilitate plant operation. The ±50 psi has been accounted for in the analyses. TS 3.2.5 is revised accordingly.

c. The low pressurizer-pressure safety-injection setpoint and time delay have been changed to 1665 psia and 30 seconds from 1760 psia and 27 seconds respectively. This pressure change will preclude an unnecessary actuation of safety injection after a normal reactor trip, because RCS pressure gets very close to the 1760 psia setpoint. The corresponding times are changed to facilitate surveillance activities. The revised LOCA analysis includes these changes. TS Table 3.3-4 is revised accordingly.

d. The analysis-values of the emergency core cooling pump performance characteristics have been changed to the "as built" characteristics. These injection performance curves were included in the LOCA as well as in the non-LOCA transient analyses. TS 4.5.2 is revised accordingly.

e. The emergency feedwater temperature and actuation time have been changed to 100°F and 75 seconds from 88°F and 60 seconds respectively. The revised values have been taken into account in the new analyses. No TS is affected by the above changes.

f. The revised LOCA analysis assumes the swelling/burst characteristics of the zirlo fuel cladding which bounds the Zircaloy cladding as well. This permits flexibility for future implementation of zirlo cladding. TS 5.3.1 is revised accordingly.

g. In the analyses of the excess feedwater flow event, credit is assumed for turbine trip and main feedwater isolation on high steam generator water level. The setpoint value has been relaxed to 94 percent from 90 percent of narrow range level in the new analysis. The staff finds the setpoint acceptable. However, no TS change was proposed at this time. The change in the analysis setpoint is margin for possible future use.

h. The constant axial offset control requirements are replaced with the ΔI LCO band defined by DNB and fuel centerline melt. This modification uses the information derived from the FIDS. TS 3.2.1 is modified accordingly. An alternate ΔI band is defined if the FIDS is not operable. Both limits are implemented in a cycle specific basis through the COLR.

2.3 Thermal and Hydraulic Analysis

The impacts of the changes in methods and parameters discussed in 2.2 above are summarized in Table 4.1 in Ref. 10. In addition, the fuel cladding thermal-performance parameters shown in Table 4.1 were determined using FROSSTEY-2, an NRC-approved code (Refs. 11 and 12). Table 4.1 reflects reasonable changes from the present UFSAR values for Seabrook.

2.4 Accident Analyses

The UFSAR accidents and transients have been reanalyzed with the new methodology and new parameter values which influence the outcome of the analyses. Such parameters are initial conditions, power distribution, reactivity coefficients, rod cluster control assembly (RCCA) insertion trip setpoints and associated time delays, component response time, and component capacities.

In addition to the uncertainties discussed in YAEC-1849P, other uncertainties accounted for in the analyses are (1) power, ±2 percent, (2) average RCS temperature, ±5.8^{0}F, and (3) pressurizer pressure, ±50 psi.

Power distribution and particularly F_Q and $F_{\Delta H}$ are discussed in YAEC-1854P (Ref. 4). The Doppler and the moderator temperature coefficient (MTC) feedback are applied in a conservative manner for the transients where they apply. The negative reactivity insertion rate following a reactor trip depends on rod insertion time. For accident analyses the time to dashpot entry is taken to be 2.4 seconds. Trip setpoints, associated time delays, component response times, and component capacities are summarized in Tables 5.0-1 to 5.0-4 of Reference 10.

2.4.1 Secondary Heat Removal Increase

The following transients result in RCS cooldown:

- Reduction in feedwater temperature
- Increase in feedwater flow
- Excessive steam flow
- Inadvertent opening of a steam generator relief valve
- Steam piping break.

The above are ANS Condition II events except for the steam piping break which is a Condition IV event. The reduction in feedwater temperature event is bounded by the increase in feedwater flow event, and the opening of a steam relief valve event is bounded by a steam pipe break event. Therefore, these two events are not reanalyzed.

2.4.1.1 Increase in Feedwater Flow

Excessive feedwater flow could increase power by decreasing RCS temperature. The overpressure-ΔT trip will protect against decreasing below the minimum departure-from-nucleate-boiling ratio (MDNBR). Excessive feedwater will activate the steam generator (SG) high water level trip which, in addition to a reactor and turbine trip, will activate feedwater isolation.

This transient is analyzed using the RETRAN code as described in YAEC-1856P (Ref. 5). Two cases (1) hot zero power, and (2) full power with the reactor in automatic control were considered. The following assumptions were made:

- At full power, an increase of 187 percent of feedwater to one steam generator due to a control valve malfunction and 200 percent at zero power
- Feedwater temperature is assumed to be at 100°F

- No credit is taken for the RCS and SG heat capacity or that the high-high SG water level trip will activate.

The results show that the DNBR does not drop below the safety analysis limit and, thus, the ability of the primary coolant to remove heat from the fuel rods is not compromised.

2.4.1.2. Increase in Secondary Steam Flow

This transient will cause a power mismatch if it is beyond a 10 percent step-increase or a 5 percent/minute ramp from 15 percent to full power. Loading rates in excess of these values will activate one of the following reactor trips: overpower-ΔT, overtemperature-ΔT, or the power range high neutron flux.

This transient is analyzed using RETRAN which simulates neutron kinetics, RCS, pressurizer with relief valves, safety valves and spray, SG and SG safety valves, and the feedwater system. This analysis is limited to the 10 percent step increase to demonstrate that the control system functions for these changes are still valid. The 10 percent step power increase bounds the 5 percent/minute ramp power increase. The analysis was performed assuming the permutations of manual and automatic reactor control with the most positive and the most negative MTC.

The results show that MDNBR remains well above the safety limit. Therefore, the 10 percent step power increase and the 5 percent/minute power ramp, are within the acceptance criteria.

2.4.1.3 Main Steam Line Break

This transient bounds the SG safety valve opening, thus, inadvertent safety valve opening or minor steam pipe breaks are not discussed.

The main steam line break (MSLB) results in excessive steam loss, RCS cooldown, reduction in pressure, and positive reactivity insertion. Assuming that the most reactive RCCA is stuck out, the core will return to power. Eventually, the reactor will be shutdown from boron injection.

The MSLB is analyzed using RETRAN and point kinetics with weighing factors derived from the STAR methodology (Ref. 13). Ref. 13 also discusses the overall reactivity insertion, the uncertainties applied to each component, peaking factors (from SIMULATE-3, Ref. 14) and DNBR calculations using VIPRE-01 with the Bowring WSC-2 CHF correlation.

Existing sensitivity studies led to the following conservative assumptions:

- End of life (EOL) shutdown margin at no load, with the most reactive RCCA stuck out

- Negative MTC corresponding to EOL, RCCA stuck out

- Offsite power available (leads to the most limiting MSLB)

- Minimum safety injection flow

- Blowdown of the three intact SGs is terminated by main steam isolation valve (MSIV) closure

- For the MDNBR calculation, the peaking factor corresponding to RCCA stuck out is assumed, and

- A high emergency-feedwater-flow with enthalpy of 50 BTU/lbm flowing to the broken loop, maximizing RCS cooldown.

Reactor protection is provided by the following functions:

- Safety injection activated from any of the following signals: low pressurizer pressure, high containment pressure, or low steam line pressure

- Overpower reactor trip (flux and overpower-ΔT)

- Main feedwater line isolation, and

- Main steam isolation on high containment pressure.

The MSLB is an ANS Condition IV event. Thus, some calculated fuel damage could be expected.

The results of the analysis shows that (1) the reactor will attain criticality at a power level less than the nominal full power, and (2) the MDNBR will remain greater than the safety limit.

2.4.2 Secondary Heat Removal, Decrease

A number of transients could result in a reduction of secondary system heat removal from the RCS. Those transients are loss of external load, turbine trip, inadvertent closure of the MSIVs, loss of condenser vacuum, loss of emergency AC power, loss of feedwater flow, or feedwater system pipe break. (Seabrook is not equipped with a steam pressure regulator). Break of a major feedwater pipe is considered to be an ANS Condition IV event, all others are considered category II. Loss of external load, MSIV closure, and loss of condenser vacuum are bounded by the turbine trip. Therefore, only turbine trip, loss of nonemergency AC power, loss of normal feedwater flow, and feedwater system pipe break, have been reanalyzed.

2.4.2.1 Turbine Trip

A turbine trip could result from either generator or transformer electrical faults, low condenser vacuum, loss of lubrication, thrust bearing failure, turbine overspeed, main steam reheat high level, or manual trip. A turbine trip is classified as ANS Condition II.

The turbine trip is analyzed using the RETRAN program, which simulates neutron kinetics, RCS, pressurizer pressure with power operated relief valves, safety valves and spray, SGs, and SG safety valves. The analysis was carried out assuming maximum power, minimum operating pressure, minimum reactivity feedback (positive MTC and minimum negative Doppler) or maximum reactivity feedback, (negative MTC and most negative Doppler), reactor control at manual, no credit for steam dump (except through safety valves), credit taken for PORVs and pressurizer spray or no credit for PORVs and pressurizer spray, loss of feedwater flow, and reactor trip when the first protection system trip setpoint is reached.

For the above cases, the results show that the reactor will trip on high pressurizer pressure and the MDNBR will remain well above the safety limit. The maximum reactor pressure will stay within design limits.

The transients initiated by main steam isolation valve closure and loss of condenser vacuum which result in a turbine trip, are covered in the above.

2.4.2.2 Loss of Nonemergency AC Power to Station Auxiliaries

Loss of nonemergency AC power may result from loss of offsite power which results in a turbine trip or from loss of the onsite AC distribution system. Upon loss of AC power the reactor will trip due to loss of power to the control rod holding coils, turbine trip, or flow coastdown. Following reactor trip, the vital instrumentation will be supplied from emergency DC power, the steam pressure will rise to the relief valve setpoint, the primary pressure will reach the PORV or the safety valve setpoints and the emergency diesel generators will start to provide emergency power. The emergency feedwater pumps will start on low-low steam generator level (or on a safety injection signal or by manual actuation). The motor driven pumps will be powered from the emergency diesels, and the steam driven pumps will be connected to the secondary and exhaust to the atmosphere. RCS natural circulation is adequate for the removal of decay heat.

This transient is analyzed using the RETRAN code, which simulates the RCS response, core neutron kinetics, the pressurizer, the SGs and the feedwater system. Conservative assumptions are made regarding initial power (102 percent), residual heat generation rate (long term operation), and heat transfer coefficients. In addition, the most positive MTC is used, the RCS T_{ave} is assumed to be high, the PORVs are assumed not to function, secondary pressure is relieved through the safety valves, only one emergency feedwater pump functions, and the reactor is tripped on low-low SG level. The results show that natural circulation will be established and that MDNBR will remain above the safety limit.2.4.2.3 Loss of Normal Feedwater Flow

Loss of normal feedwater can result from a pump failure, a valve malfunction, or a loss of offsite power. Following loss of normal feedwater, steam pressure will rise and pressure will be relieved through the steam generator PORVs or the secondary safety valves. The reactor will be tripped on low-low SG water level. Emergency feedwater will be initiated as discussed in the previous section.

The loss of normal feedwater transient is analyzed using the RETRAN code simulating core neutron kinetics, RCS, the pressurizer (pressure and level), the SGs, and the feedwater system. The same conservative assumptions are made as in the analysis of the loss of AC power above.

The results show that the SG level will fall to the low-low reactor trip setpoint with 75 seconds to trip the reactor. No relief from the pressurizer will be required and the plant will approach a stabilized condition in a time frame in excess of 10 minutes. The MDNBR remains well above the safety limit and the maximum pressure is well within the design safety limits.

2.4.2.4 Feedwater System Pipe Break

For this case, a double-ended rupture in a feedwater line is assumed between the SG and a check valve, located about 60 feet outside from the containment wall upstream from the feedwater pump (see UFSAR Fig. 10.4-9 Sheet 1). Complete loss of normal feedwater and emergency feedwater through the break is assumed. (The case of a break upstream of the check valve is bounded by the loss of feedwater or loss of AC power to the station). Depending on the size of the break, the RCS could heat up and pressurize or cooldown and experience lower pressure. A reactor trip can be caused by high pressurizer pressure, overtemperature-ΔT, low-low SG water level, any safety injection signal, or emergency feedwater activation.

This transient is analyzed using the RETRAN code using worst-case conditions, derived from sensitivity studies reported in WCAP-9320 (Ref. 15). The analysis assumptions include: initial power 102 percent, RCS temperature 5.8°F above nominal, pressurizer pressure 50 psi above nominal, pressurizer level 170 ft^3 above nominal, SG level nominal plus 5 percent, the least negative Doppler and most positive MTC, no PORVs or pressurizer spray, no credit for charging and letdown, worst double-ended break, turbine trip is assumed at the time of the break, no credit for atmospheric dump, reactor trip and EFW actuation on SG low-low level, safety injection is credited on low main steam line pressure, minimum ECCS pump performance, maximum ECCS water temperature (100°F) and only one EFW train is assumed operable (single failure).

The results show that the RCS will initially heatup prior to reactor trip which is followed by a cooldown and subsequent heatup due to MSIV closure (terminating the blowdown of the intact SGs) which lasts until the EFW heat removal capability is sufficient to remove decay and RCS pump heat. The MDNDR remains above the safety limit, thus, the EFW system response is adequate to assure decay heat removal, prevent overpressurization and prevent core uncovery.

2.4.3 Decrease in Reactor Coolant Flow Rate

Partial or total loss of RCS flow and coolant pump shaft break or seizure are events under this heading. The shaft break is bounded by shaft seizure and is considered an ANS Condition IV event. Partial and total loss of RCS flow are considered ANS Condition II and III respectively.

2.4.3.1 Complete Loss of RCS Flow

This event may result from loss of electrical power supply to all reactor coolant pumps. Loss of flow will result in rapid increase of coolant and fuel temperature if the reactor is not tripped. Normal power to the pumps is supplied from the station generator. If there is a turbine trip the generator breaker will open and the pumps are switched to offsite power. There are two reactor trip signals to protect against total loss of RCS flow: low RCS flow and RCS pump undervoltage or underfrequency. Reactor protection from underfrequency events is discussed in WCAP-8424, Revision 1, (Ref. 16). This event is analyzed using the RETRAN code, to calculate loop flow, RCS pressure, and coolant temperature as a function of time. The CHIC-KIN (Ref. 17) code is then used to calculate the heat flux transient and finally VIPRE-01 is used to calculate the MDNBR.

Conservative assumptions are made regarding initial conditions for, flow, power, pressure and coolant temperature. The maximum Doppler and positive MTC (to allow for uncertainties) are used to maximize power production. Finally, a slightly bottom skewed power distribution and a 2.4 seconds rod drop (to the dashpot) are assumed.

The results show that the MDNBR remains above the safety limit. Following pump coastdown and reactor trip, natural circulation will be established which will be able to remove decay heat.

With respect to MDNBR, the complete loss of RCS flow bounds the case of partial loss of RCS flow, thus, partial RCS flow is not discussed separately.

2.4.3.2 Reactor Coolant Pump Shaft Seizure

This accident results from an instantaneous seizure of a reactor coolant pump rotor. The resulting low loop-flow will trip the reactor, and the reduced heat transfer will increase the pressurizer level and activate the PORVs (and possibly lifting the pressurizer safety valves). This is classified as an ANS Condition IV event.

This transient is analyzed using (1) the RETRAN code to calculate primary flows and the primary pressure transient, (2) the CHIC-IN kinetics code is used for the estimation of the hot rod power distribution and power level, and (3) the VIPRE-01 code to calculate DNBR.

The results show that the peak RCS pressure and the peak clad temperature are bounded by the locked rotor with loss of offsite power transient. The peak pressure is well below the 110 percent of design value (2750 psia) and MDNBR remains bounded by the UFSAR value.

2.4.3.3 Reactor Coolant Pump Shaft Seizure with Loss of Offsite Power

This is the most limiting of the locked rotor and broken shaft accidents. In this event, following shaft break/locked rotor the remaining three RCS pumps continue to operate. Analysis of the electrical supply to the 13.8 KV bus which supplies two of the four RCS pumps indicates that the locked rotor/broken shaft will not affect power supply. Nevertheless, it is assumed that when a rotor locks and reactor trips on low loop-flow, the turbine trips, then loss of offsite power causes the three remaining RCPs to coast down.

The analysis is performed using the RETRAN code in conjunction with the CHIC-KIN and VIPRE-01. A conservative valve of the maximum pressure is estimated by assuming initial values of maximum power, pressure and coolant temperature. DNBR is estimated using the RTDP nominal valves and associated uncertainties.

The results show that the maximum pressure will remain below 110 percent of the design pressure of 2,500 psia. The MDNBR will be below the safety limit for about 8 percent of the fuel rods, which is unchanged from the value that is estimated in the current revision of the UFSAR. The peak clad temperature is about 1100°F, thereby, assuring core fuel integrity and no loss of cooling capability.

2.4.4 Reactivity and Power Distribution Anomalies

Reactivity and power distribution anomalies could be caused by control rod motion, boron concentration changes, addition of cold water to the RCS, control rod misalignment, or fuel assembly mislocation. The following events are reanalyzed: uncontrolled RCCA bank withdrawal from subcritical or low power, uncontrolled RCCA withdrawal at power, RCCA misalignment, and a spectrum of RCCA ejections. The remaining events, i.e., startup of an inactive coolant pump at an incorrect temperature and operation with a fuel assembly loaded into an improper position, are either precluded by technical specification or can be detected without consequence.

2.4.4.1 Uncontrolled RCCA Withdrawal from Subcritical or Hot Zero Power.

This event can be caused by withdrawal of RCCAs which result in a large reactivity insertion and a power excursion from subcritical or hot zero power. (RCCA withdrawal from power is discussed separately). The maximum reactivity insertion rate occurs with the simultaneous withdrawal of two sequential RCCAs at maximum speed and combined worth.

The rapid neutron power rise will be terminated by the Doppler reactivity coefficient. This provides some delay action for the protective system which will respond with actuation of the source range, the intermediate range, the power range (low setting) or the power range high neutron flux reactor trip, (high setting), or the nuclear flux rate reactor trip. In addition, control rod stops on high intermediate range flux level and high power range flux will interrupt rod withdrawal.

This transient was analyzed in three stages: first the core power transient was calculated, then the core heat transfer and finally the MDNBR. The SIMULATE-3 (Ref. 14) the CHIC-KIN (Ref. 17) and the VIPRE-01 (Ref. 6) codes respectively were used for the above three stages. In the analyses, the following assumptions and initial conditions are used: low value of the Doppler coefficient, low MTC for the last two stages, reactor at hot zero power, reactor trip is assumed to be initiated by the power range (low setting) with the most adverse combination of instrument and setpoint errors and signal delay for the RCCA scram, the highest worth RCCA is assumed in the stuck out position, reactivity insertion is greater than the maximum rate of two sequential RCCA withdrawals at maximum speed, the most limiting axial and radial power shapes are assumed, and only two RCS pumps are assumed operating.

The results show that the total energy release is relatively small, and the MDNBR remains well above the safety analysis limit. Following this transient, the reactor can be cooled further following normal plant shutdown procedures.

2.4.4.2 Uncontrolled RCCA Withdrawal at Power

Power increase will produce a power generation to power removal mismatch. The following reactor protection features will prevent core damage by tripping the reactor: power range neutron high flux, overtemperature-ΔT, overpower-ΔT, high pressurizer pressure and high pressurizer water level. In addition, the high neutron flux, the overtemperature-ΔT, and overpower-ΔT trips will also activate RCCA withdrawal blocks, preventing a transient due to control rod reactivity insertion.

This transient is analyzed using the RETRAN code which simulates the RCS, neutron kinetics, pressurizer and pressurizer PORVs, safety valves, and spray, SGs and SG safety valves. The code calculates plant variables including RCS pressure, temperature, and reactor power level. To assure conservative results, initial conditions assumed maximum power and RCS temperature. Cases were run with minimum and maximum reactivity feedback accounting for Doppler and MTC.

The results show that the reactor will trip on high neutron flux and the MDNBR will remain above the safety analysis limit.

2.4.4.3 RCCA Misalignment (Misoperation)

This transient can be caused by a dropped RCCA, a dropped bank, a statically misaligned RCCA or RCCA withdrawal. Reactor protection for this event is the same as listed in Section 2.4.4.2. In addition a dropped or misoperating RCCA can be detected by the nuclear instrumentation, core exit thermocouples due to power asymmetry, rod bottom signals, rod deviation alarms, and rod position indications. Multiple electrical and operational faults and errors are required to cause prolonged operation with any of the above. The dropped RCCA assemblies, dropped assembly banks, and statically misaligned assembly events are classified as ANS Condition II, i.e., incidents of moderate frequency. However, prolonged operation with a single RCCA withdrawn is considered an ANS Condition III event, thus, limited fuel damage can be tolerated.

Analysis of any of the RCCA misoperation transients is performed using the RETRAN code which simulates kinetics, the RCS, the pressurizer and pressurizer PORVs, relief valve, and spray, the SGs and the SG safety valves. Following the RETRAN analysis, VIPRE-01 is used to estimate MDNBR.

The results show that only for the case of a single rod withdrawal, will the MDNBR fall below the safety analysis limit. The estimated number of affected fuel rods will be limited to about 5 percent. For this level of affected rods the radiological consequences are acceptable.

2.4.4.4 Spectrum of RCCA Ejection Accidents

This accident can be caused by failure of the control rod pressure housing resulting in RCCA ejection. Such an event will cause power to increase but more importantly will cause localized power maldistribution which can lead to localized fuel damage. At Seabrook, the design of the rod housing and the mode of operation, i.e., small RCCA insertion for load following, would either preclude or minimize the results of a rod ejection. This event is classified as an ANS Condition IV event.

This event is analyzed using the CHIC-KIN and the STAR codes to estimate power distribution and Doppler reactivity from the local power distribution. MDNBR analysis is performed using the VIPRE-01 code.

The results show that in the worst case MDNBR will involve less than 10 percent of the rods. The resulting releases from the combination of rod ejection and the ensuing LOCA is the same as that analyzed in the UFSAR.

2.4.5 Increase in Reactor Coolant Inventory

Increases in the reactor coolant inventory can be caused by either, inadvertent operation of the emergency core cooling system (ECCS) during power operation or malfunction of the chemical and volume control system (CVCS) which increases the RCS inventory.

2.4.5.1 Inadvertent ECCS injection During Power Operation

UFSAR documented results show that ECCS injection during power operation does not present a problem. If the reactor does not trip immediately the low pressure reactor trip will be activated. The DNBR is never lower than its original value, thus, this event need not be reanalyzed.

2.4.5.2 CVCS Malfunction to Increase RCS Inventory

Transients due to CVCS malfunction which increase RCS inventory are classified into three categories, i.e., injection of water with boron concentration greater than, equal to, or lower than in the RCS. The consequences are bounded by the case of water injection with boron concentration lower than that in the RCS. The UFSAR studies demonstrated that such a malfunction will not result in a significant power or temperature transient, therefore, this event has not been reanalyzed.

2.4.6 Decrease in Reactor Coolant Inventory

A decrease in RCS inventory can be caused by inadvertent opening of a PORV or safety valve, a RCS instrument or other line break, SG tube rupture, or LOCA through pipe breaks within the RCS pressure boundary.

2.4.6.1 Inadvertent Opening of a PORV or Safety Valve

Accidental opening of a safety valve can discharge about twice the amount of steam than a PORV, therefore, the event consequences of a safety valve opening bound those of a PORV opening. Pressurizer safety valves are sized such as to discharge twice the amount of steam mass at operating reactor pressure.

The event causes an initial RCS depressurization until the pressure reaches the hot leg saturation pressure, then the pressure decrease slows considerably. The reactor control system will increase power due to MTC feedback. The RCS T_{ave} will increase until pressurizer high level trips the reactor. The reactor may be tripped by the following reactor protection system signals: overtemperature-ΔT or low pressurizer pressure.

This event has been analyzed using the RETRAN code which simulates reactor kinetics, RCS, pressurizer and pressurizer PORVs, safety valves, level, and spray, SGs and SG relief and safety valves. The assumed initial conditions of maximum power and RCS temperature and minimum pressure minimize initial DNBR margin. In addition the most positive MTC and minimum Doppler coefficients are assumed to minimize reactivity feedback. Assuming that reactor control is in manual and no action is taken, the reactor will trip. Reactor recovery is similar to a small LOCA.

Results of the analysis show that initially the DNBR will decrease but will remain well above the safety analysis limit.

2.4.6.2 SG Tube Rupture

This event has been reanalyzed as part of a separate licensing action described in YAEC-1698 (Ref. 18) which has been approved by the staff. Therefore, it is not reviewed here.

3.0 <u>LOSS-OF-COOLANT ACCIDENTS</u>

North Atlantic's submittal (Ref. 1) included as supporting information, a report titled "Seabrook Station Fuel Upgrade Program LOCA Safety Analysis Report". This report presents a reanalysis of the large break and small break LOCA accidents. This report discusses ANS Condition III and IV events, the LOCA acceptance criteria to satisfy the requirements of 10 CFR 50.46, large LOCA phenomenology, reactor protection system response, the evaluation model, and the computer programs used and their application and compliance with the requirements of Appendix K to 10 CFR 50 and Item II.K.3.5. of NUREG-0737. The report also provides the initial conditions and the results of the Seabrook large and small break LOCA reanalyses.

3.1 Evaluation

3.1.1 Large Break LOCA

The analysis and the report organization follow the LOCA phenomenology, namely the three phases of the transient - blowdown, refill, and reflood. For each of the phases, the RCS thermal-hydraulic transient, the containment pressure and temperature transients, and the core fuel and fuel cladding hot rod transient are analyzed.

The RCS thermal-hydraulic transient is computed using the SATAN-VI code (Refs. 19 - 22) including the RCS pressure, enthalpy, coolant density, mass and energy flow rates in the RCS, and SG energy transfer between primary and secondary. In addition, SATAN-VI computes mass and energy release rates to the containment to be input to the COCO code, (Ref. 23), and the interface for the refill and reflood phases.

The WREFLOOD code (Ref. 24) calculates recovery time and the mass and energy from the break during the LOCA reflood phase. The COCO code computes the containment pressure and is interactively linked to WREFLOOD. Data from SATAN-VI are also used by the LOCBART code (Ref 25) to calculate core average conditions for use by the BASH code (Ref. 26) which in turn computes the core and RCS thermal-hydraulic parameters during the reflood phase.

During the refill phase, the accumulator conditions and the injection flow are provided to BASH by WREFLOOD. LOCBART provides the enthalpy, pressure and the details of the fuel rod during refill. During reflood and refill a dynamic interaction between core thermal-hydraulics and system behavior takes place. Loop flows and pressure drops are a function of core exit flow and the new entrainment rate is fed back into the loop calculation. This dynamic interaction is accomplished by continuous data exchange between BASH (the loop code) and BART, the reflood code, (Ref. 27). Fuel rod parameters in the reflood and refill phases are again calculated by LOCBART which incorporates the FLECHT empirical correlation for the calculation of the heat transfer coefficients appropriate for the actual flow and heat transfer regimes experienced by the fuel rods.

The initial RCS, core, and containment conditions at LOCA initiation were chosen conservatively on the basis of Westinghouse sensitivity studies (Refs. 28 - 30). In addition the requirements of Appendix K to 10 CFR 50 regarding specific model features are met by model selection which provides significant conservatism in the analysis. The decay heat generated during the transient also is calculated conservatively consistent with the requirements of Appendix K to 10 CFR 50. For Seabrook, it was found that minimum ECCS flow results in the highest peak clad temperature (PCT) for the limiting break, i.e., the double-ended cold-leg guillotine break. From the range of the Moody discharge coefficients, C_D, used (0.4, 0.6 and 0.8), the maximum PCT of 1889°F was estimated for C_D=0.6. A 5°F penalty was added to account for the RTD bypass elimination, yielding 1894°F, which is within the acceptance criteria of 2200°F. The total core metal-water reaction is less than 1.0 percent, as required by 10 CFR 50.46. The transient is terminated while core geometry is amenable to cooling.

3.1.2 Small Break LOCA

A small-break LOCA is defined as the rupture of a RCS pipe with cross sectional area of less than 1.0 ft^2. The normal charging system is not sufficient to maintain pressurizer pressure and level; thus, the reactor will trip on low pressurizer pressure. Loss of offsite power is assumed to coincide with the small-break LOCA. Safety injection with borated water will be initiated. The boron concentration is sufficient to assure that the post-LOCA core would remain subcritical.

There is sufficient ECCS water injection to assure that excessive clad temperatures will not occur. The most limiting active failure is the one which minimizes ECCS flow. This has been determined to be the loss of a power train which results in the loss of a complete ECCS train. The safety injection pump, the centrifugal charging pump and the residual heat removal pump performance curves were assumed degraded by 5 percent, 5 percent and 8.75 percent, respectively.

The small-break LOCA is analyzed using the NOTRUMP code (Refs. 31 - 33) to estimate the transient depressurization and mass and energy flow through the break. Clad thermal analyses are performed with the LOCTA-IV code (Ref. 34), using the NOTRUMP output for RCS pressure, fuel rod power history, core uncovery, steam flow, and mixture heights as boundary conditions. Fuel Rod axial power distribution is biased upward to maximize vapor superheating and fuel rod heat generation in the uncovered elevations.

The methodology in this analysis complies with the requirements of Appendix K to 10 CFR 50, and with the revisions of section II.K.3.30 of NUREG-0737 (Ref. 35), as shown through sensitivity studies and generic analyses.

The results show that the limiting small-break LOCA with respect to PCT is the 4-inch break which yields a PCT of 1082°F. Adding 8°F penalty for RTD bypass elimination the final PCT value is 1090°F PCT. In addition, the metal-water reaction is less than 1.0 percent; thus, the results are within the 10 CFR 50.46 criteria. Therefore, the LOCA analyses for Seabrook are acceptable.

4.0 CORE OPERATING LIMITS REPORT

North Atlantic proposed a revised core operating limits report (COLR) to be used for cycle 5 and subsequent cycles. Technical specification 6.8.1.6 provides the format and the specific parameters which should be included in the COLR report.

The staff has reviewed the proposed COLR, and finds that it complies with the TS 6.8.1.6 format. In addition, the parameter values have been reviewed and approved by the staff. Therefore, we find the proposed COLR for the operation of Seabrook for cycle 5 and subsequent cycles to be acceptable.

5.0 EVALUATION

The staff has reviewed the reanalyses of accidents and transients for Seabrook submitted in support of the proposed changes to the Seabrook TS. The methodologies that were used for the reanalyses are described in Refs. 3 - 6.

The staff finds that the analyses of transients and accidents discussed in this safety evaluation were performed using methods approved previously. Based on our review, we have determined that: (1) the initial conditions and assumptions used in the reanalyses are such as to lead to conservative values of the parameters of interest, and (2) the results are within the acceptance criteria of the standard review plan and applicable regulations. The staff finds the results to be acceptable, and therefore, concludes that the proposed changes to the Seabrook TS, except the changes proposed to TS 3.1.1.3, are acceptable.

The proposed changes to TS 3.1.1.3 would permit operation of Seabrook with a positive MTC. The staff has requested North Atlantic to submit certain additional information that is necessary for the staff to consider further the acceptability of operating Seabrook with a positive MTC.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and was published in the Federal Register on October 3, 1994 (59 FR 50259). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

7.0 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.

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Date: November 23, 1994

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTH ATLANTIC ENERGY SERVICE CORPORATION, ET AL

DOCKET NO. 50-443

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 33 to Facility Operating License No. NPF-86, issued to North Atlantic Energy Service Corporation (the licensee), which revised the Technical Specifications for operation of the Seabrook Station, Unit No. 1 (Seabrook) located in Rockingham County, New Hampshire. The amendment is effective as of the date of its issuance, to be implemented before startup from the fourth refueling outage.

The amendment modifies the Seabrook Technical Specifications (TS) to permit operation of the Seabrook core with an expanded axial flux difference band (wide-band operation) from that currently permitted. Other TS changes allow for fuel design enhancements. Wide-band operation is based on information derived from the fixed in-core detector system (FIDS). The core design enhancements are based on methodologies described in Yankee Atomic Electric Company reports YAEC-1849P, YAEC-1854P, and YAEC-1856P which were approved previously for use at Seabrook. Additionally, the licensee supported the proposed technical specification changes with reanalyses of the Updated Final Safety Analysis Report Chapter 15 accidents and transients, documented in YAEC-1871, and a Westinghouse Electric Corporation loss-of-coolant-accident reanalysis. These supporting documents and a revised Core Operating Limits Report were submitted with the application for amendment.

The licensee proposed certain changes to TS 3.1.1.3 that would permit operation with a positive moderator temperature coefficient. The Commission has not yet determined the acceptability of this proposed change pending submission of additional information from the licensee. Therefore, the proposed change to TS 3.1.1.3 is not implemented by this amendment.
This amendment affects TS Sections 3.1.3.4, 3.2.1, 3.2.2, 3.2.3, 3.2.4, 3.2.5, 3.3.3.2, 4.2.1, 4.2.2, 4.2.5, 4.5.2, 5.3, and 6.8.1, Figure 2.1-1, and Tables 2.2-1, 3.3-4, and 4.3-1.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on January 18, 1994 (59 FR 2632). No request for hearing or petition for leave to intervene was filed following this notice.

For further details with respect to this action see (1) the application for amendment dated November 23, 1993, as supplemented by letter dated August 15, 1994, (2) Amendment No. 33 to License No. NPF-86, and (3) the Commission's related Safety Evaluation, and (4) the Commissions Environmental Assessment dated September 27, 1994. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Exeter Public Library, 47 Front Street, Exeter, NH 03833.

Dated at Rockville, Maryland this 23rd day of November 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

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