March 20, 1989

Docket No. 50-336

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Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 68360)

The Commission has issued the enclosed Amendment No. 139 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2, in response to your applications dated November 15, 1988 and February 1, 1989.

This amendment allows operation of Millstone Unit 2 for Cycle 10. The changes to the Technical Specifications reflect a revised safety analysis that includes the use of fuel designed and fabricated by Advanced Nuclear Fuels Corporation. Fuel designed and fabricated by ANF has not been previously utilized for Millstone Unit 2. The changes to the Technical Specifications also reflect the effects of reduced reactor coolant flow from 340,000 to 325,000 gpm.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

original signed by

Guy S. Vissing, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 139 to DPR-65
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDI-4	PM:PDI-4	PD:PDI-4	OGC
SNorris	GVissing:cb	JStolz	
03/9/89	03/10/89	03/17/89	03/13/89

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Mr. Edward J. Mroczka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 2

cc:

Gerald Garfield, Esquire
Day, Berry and Howard
Counselors at Law
City Place
Hartford, Connecticut 06103-3499

R. M. Kacich, Manager
Generation Facilities Licensing
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

W. D. Romberg, Vice President
Nuclear Operations
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

D. O. Nordquist
Manager of Quality Assurance
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Kevin McCarthy, Director
Radiation Control Unit
Department of Environmental Protection
State Office Building
Hartford, Connecticut 06106

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

Bradford S. Chase, Under Secretary
Energy Division
Office of Policy and Management
80 Washington Street
Hartford, Connecticut 06106

First Selectmen
Town of Waterford
Hall of Records
200 Boston Post Road
Waterford, Connecticut 06385

S. E. Scace, Station Superintendent
Millstone Nuclear Power Station
Northeast Nuclear Energy Company
Post Office Box 128
Waterford, Connecticut 06385

W. J. Raymond, Resident Inspector
Millstone Nuclear Power Station
c/o U. S. Nuclear Regulatory Commission
Post Office Box 811
Niantic, Connecticut 06357

J. S. Keenan, Unit Superintendent
Millstone Unit No. 2
Northeast Nuclear Energy Company
Post Office Box 128
Waterford. Connecticut 06385

Charles Brinkman, Manager
Washington Nuclear Operations
C-E Power Systems
Combustion Engineering, Inc.
12300 Twinbrook Pkwy
Suite 330
Rockville, Maryland 20852

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

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
License No. DPR-65

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

   A. The applications for amendment by Northeast Nuclear Energy Company, et al. (the licensee), dated November 15, 1988 and February 1, 1989 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   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.

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. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 139, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, 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: March 20, 1989

ENGINEERED SAFETY FEATURE RESPONSE TIME (Continued)

performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.28 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.29 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

1.31 A RADIOLOGICAL EFFLUENT MONITORING MANUAL shall be a manual containing the site and environmental sampling and analysis programs for measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures to individuals from station operation. An OFFSITE DOSE CALCULATION MANUAL shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints. Requirements of the REMODCM are provided in Specification 6.16.

RADIOACTIVE WASTE TREATMENT SYSTEMS

1.33 RADIOACTIVE WASTE TREATMENT SYSTEMS are those liquid, gaseous and solid waste systems which are required to maintain control over radioactive material in order to meet the LCOs set forth in these specifications.

PURGE - PURGING

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

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) used for normal control and indication is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad YI = AY_E + B$$

1.24 Deleted.

ENCLOSURE BUILDING INTEGRITY

1.25 ENCLOSURE BUILDING INTEGRITY shall exist when:

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

1.25.2 The enclosure building filtration system is OPERABLE.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERING SAFETY FEATURE RESPONSE TIME

1.27 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

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

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

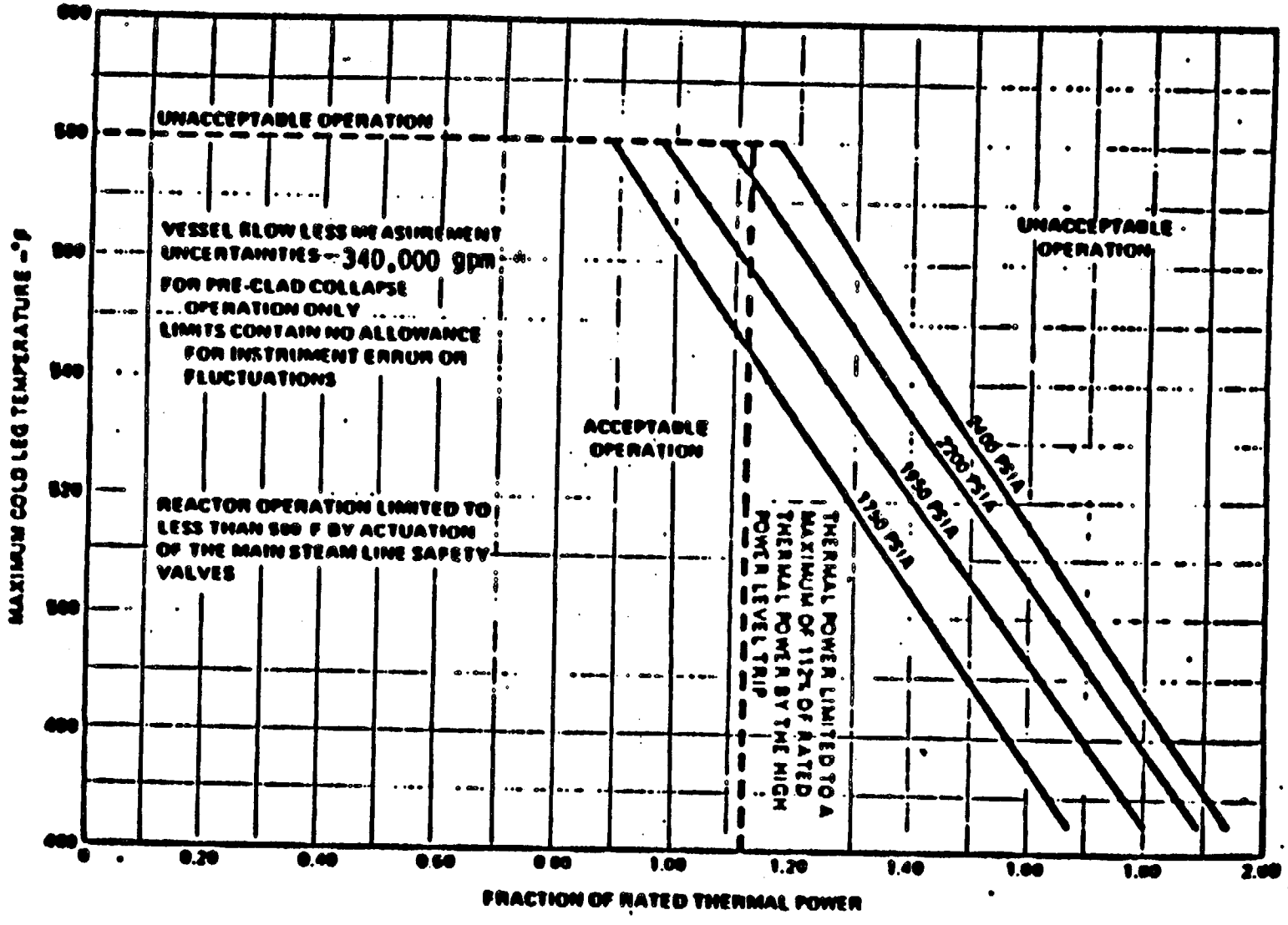

FIGURE 2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT - FOUR REACTOR COOLANT PUMPS OPERATING

*Flow reductions to 325,000 gpm are compensated for by reductions in the F_r^T limit (Specification 3.2.3)

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation 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:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level-High		
Four Reactor Coolant Pumps Operating	≤ 9.6% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER.	≤ 9.7% Above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.1% of reactor coolant with 4 pumps operating.
4. Reactor Coolant Pump Speed - Low	≥ 830 rpm	≥ 823 rpm
5. Pressurizer Pressure - High	≤ 2400 psia	≤ 2408 psia
6. Containment Pressure - High	≤ 4.75 psig	≤ 5.24 psig
7. Steam Generator Pressure - Low (2) (5)	≥ 680 psia	≥ 672 psia
8. Steam Generator Water Level - Low (5)	≥ 36.0% Water Level - each steam generator	≥ 35.2% Water Level - each steam generator
9. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

*Design Reactor Coolant flow with 4 pumps operating is the lesser of either:
a. The reactor coolant flow rate measured per specification 4.2.6.1, or
b. 340,000 gpm

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4(4).
11. Loss of Turbine--Hydraulic Fluid (3) Pressure - Low	≥ 500 psig	≥ 500 psig

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ± 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 780 psia when all CEAs are fully inserted; bypass shall be automatically removed at or above 780 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER IS ≥ 15% of RATED THERMAL POWER.

(4) Calculations of the trip setpoint includes measurements, calculational and processor uncertainties, and dynamic allowances.

(5) Each of four channels actuate on the auctioneered output of two transmitters, one from each steam generator.

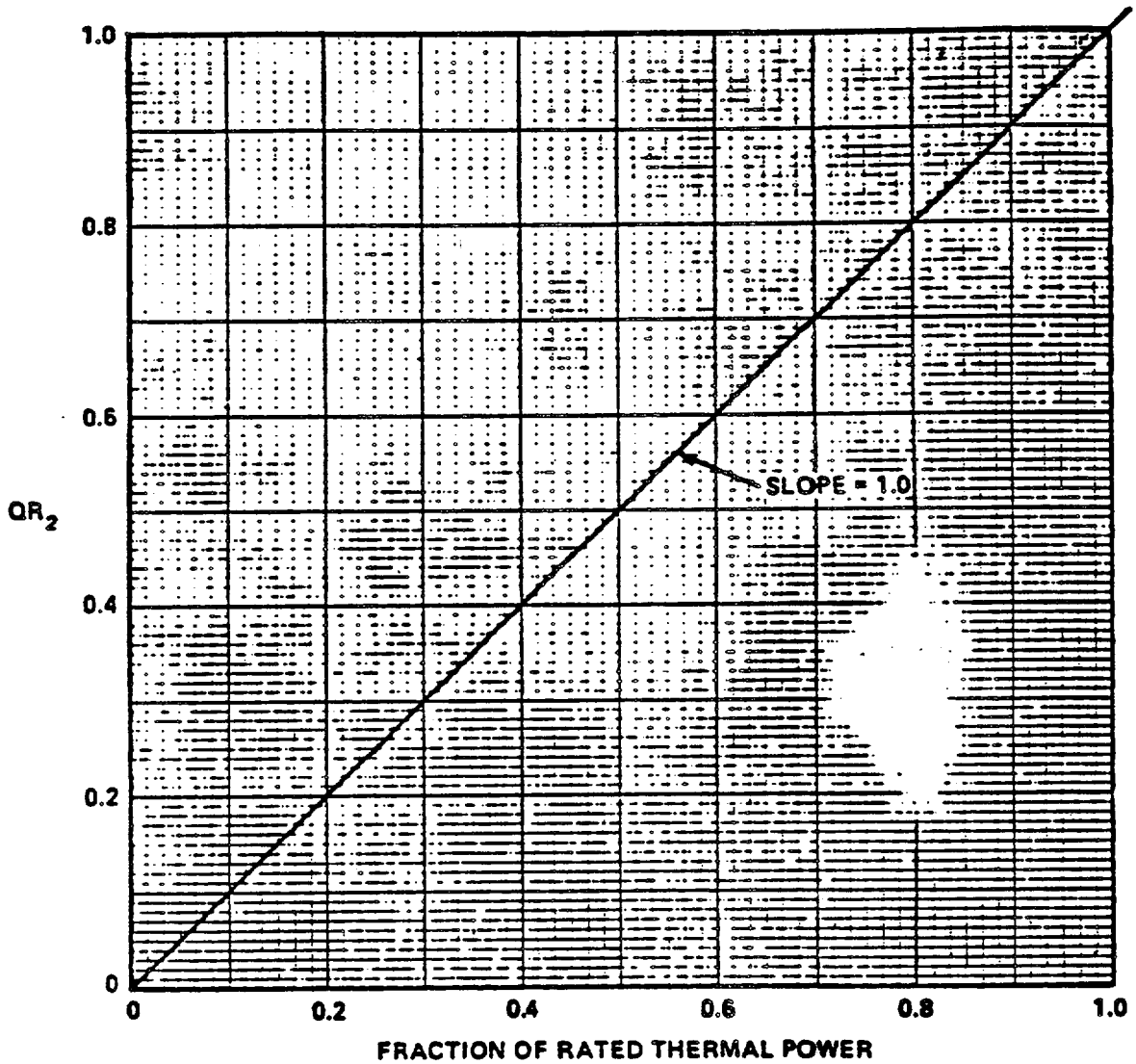

FIGURE 2.2-1

Local Power Density – High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. 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 through the XNB correlation. The XNB DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.17. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.17. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperatures is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe operation is below and to the left of these lines.

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The conditions for the Thermal Margin Safety Limit curves in figure 2.1-1 to be valid are shown on the figure.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than 1.17 and preclude the existence of flow instabilities.

2.1.2 REACTOR COOLANT SYSTEM PESSURE

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

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SET POINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed to occur for each trip used in the accident analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 9.6% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 106.6% of RATED THERMAL POWER and a minimum setpoint of 14.6% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the accident analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit

Reactor Coolant Flow-Low (Continued)

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.17 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.17 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpont for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 680 psia is sufficiently below the full-load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 22 psi in the accident analyses.

Steam Generator Water Level - Low

The Steam Generator Water Level-Low Trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 10 minutes before auxiliary feedwater is required.

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.17.

Thermal Margin/Low Pressure (Continued)

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1850 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2°F to compensate for potential temperature measurement uncertainty; and a further allowance of 72 psi to compensate for pressure measurement error, trip system processing error, and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 72 psi allowance is made up of a 22 psi pressure measurement allowance and a 50 psi time delay allowance.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuring transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these values. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Underspeed - Reactor Coolant Pumps

The Underspeed - Reactor Coolant Pumps trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant pump speed (with resulting decrease in flow) on all four reactor coolant pumps. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained above 1.17 following a 4 pump loss of flow event.

3/4.1 REACTVITY CONTROL SYSTEMS

3/4.1.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ 3.60% ΔK/K.

APPLICABILITY: MODES 1, 2*, 3 and 4

ACTION:

With the SHUTDOWN MARGIN < 3.60% Δk/k, within 15 minutes initiate and continue boration at ≥ 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the required SHUTDOWN MARGIN is reached.

SURVEILLANCE REQUIREMENT

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ 3.60% ΔK/K:

a. Immediately upon detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, the SHUTDOWN MARGIN, required by Specification 3.1.1.1, shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA.

b. When in MODES 1 OR 2, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. Prior to initial operation above 5% RATED THERMAL POWER after each refueling, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% Δk/k at least once per 31 Effective Full Power Days. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.d, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each refueling.

MODERATOR TEMPERATURE COEFFICIENT (MTC)

LIMITING CONDITION FOR OPERATION (Continued)

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.7×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.4×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.8×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENT

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparision with the predicted values.

*With $K_{eff} \geq 1.0$.

#See Special Test Exemption 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each refueling.

b. At any THERMAL POWER, within 14 EFPD after each fuel loading at equilibrium boron concentration.

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION (Continued)

3.2.1 The linear heat rate, including heat generated in the fuel, clad and moderator, shall not exceed:

a. 15.1 kw/ft when the reactor coolant flow rate measured per Specification 4.2.6.1 $\geq$ 340,000 gpm.

b. 14.5 kw/ft when the reactor coolant flow rate measured per Specification 4.2.6.1 $\geq$ 325,000 gpm and < 340,000 gpm.

APPLICABILITY: MODE 1.

ACTION:

During operation with the linear heat rate being monitored by the Incore Detector Monitoring System, comply with the following ACTION:

With the linear heat rate exceeding the limit as indicated by four or more coincident incore channels, within 15 minutes initiate corrective action to reduce the linear heat rate to less than or equal to the limit and either:

a. Restore the linear heat rate to less than or equal to the limit within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

During operation with the linear heat rate being monitored by the Excore Detector Monitoring System, comply with the following ACTIONS:

With the linear heat rate exceeding its limit, as indicated by the AXIAL SHAPE INDEX being outside of the power dependent limits on the Power Ratio Recorder, either:

a. Restore the AXIAL SHAPE INDEX to within the limits of Figure 3.2-2 within 1 hour from initially exceeding the linear heat rate limit, or

b. Be in at least HOT STANDBY within the next 4 hours.

SURVEILLANCE REQUIREMENT

4.2.1.1 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

SURVEILLANCE REQUIREMENT (Continued)

4.2.1.2 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 12 hours that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6.

b. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the allowable limits of Figure 3.2-2.

4.2.1.3 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days.

b. Have their alarm setpoint adjusted to less than or equal to the limit when the following factors are appropriately included in the setting of these alarms:

*1. Flux peaking augmentation factors as shown in Figure 4.2-1.
2. A measurement-calculational uncertainty factor of 1.07,
3. An engineering uncertainty factor of 1.03,
*4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and
5. A THERMAL POWER measurement uncertainty factor of 1.02.

*These factors are only applicable to fuel batches "A" through "L"

FIGURE 3.2-1 LEFT BLANK INTENTIONALLY

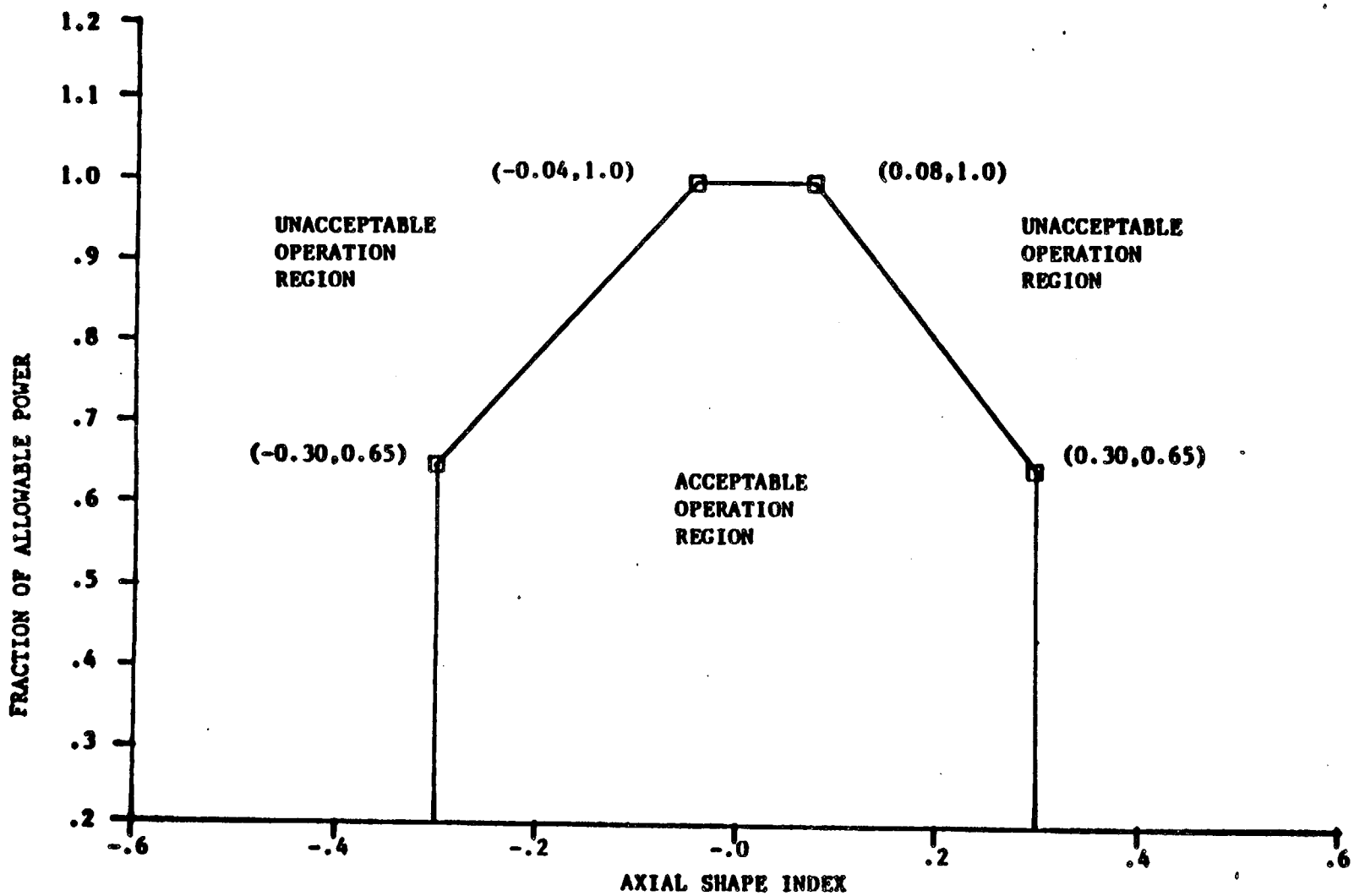

FIGURE 3.2-2 AXIAL SHAPE INDEX vs. FRACTION OF ALLOWABLE POWER LEVEL

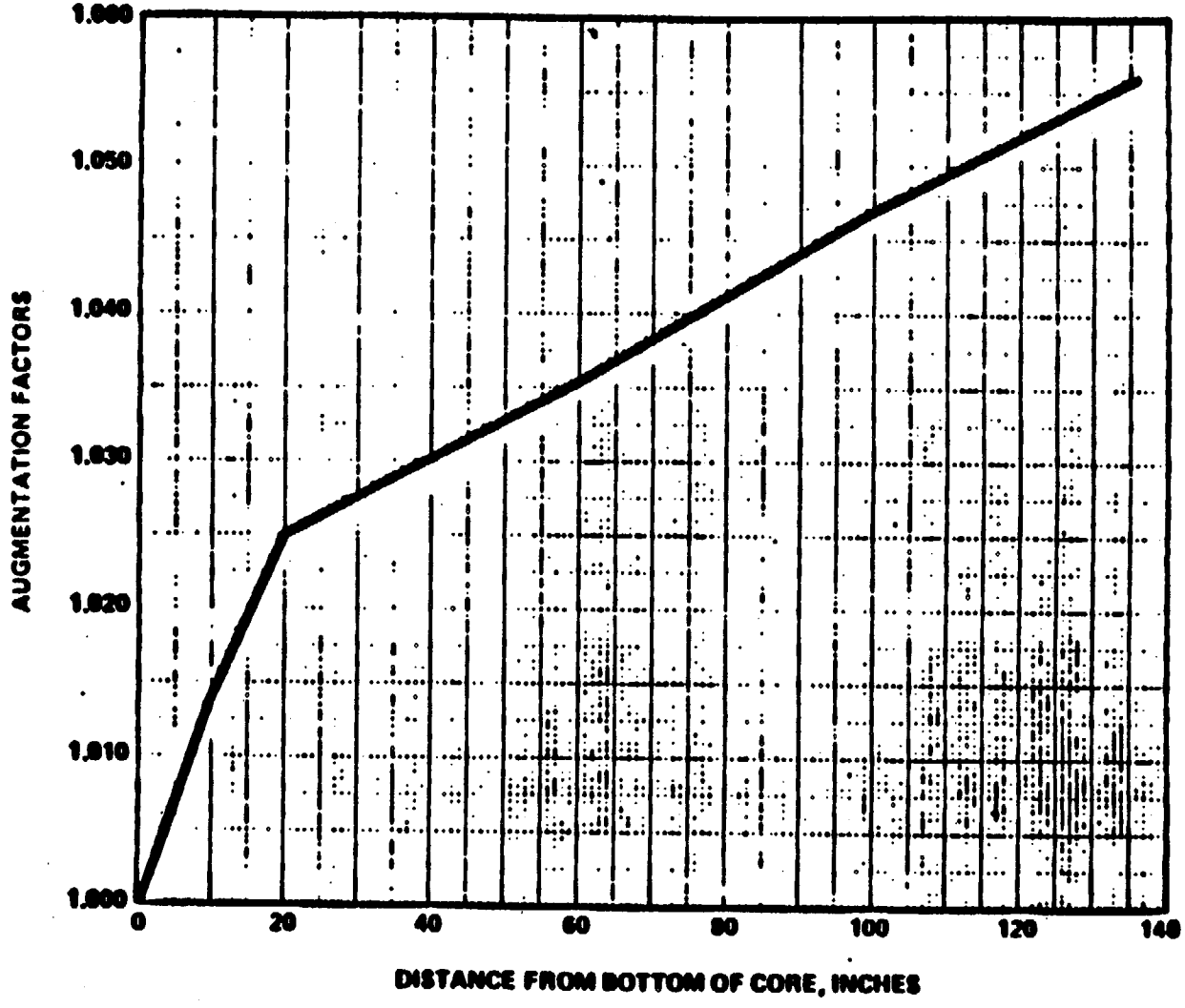

FIGURE 4.2.1 Augmentation Factor vs. Distance from Bottom of Core
(only applicable to fuel batches "A" through "L")

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TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T defined as $F_r^T = F_r\ (1+T_q)$, shall be limited to:

a. $0.90 < PF \leq 1.00$ $\quad F_r^T \leq (11.73 - PF)((1.24 \times 10^{-7} \times FL)+0.108)$

b. $0.70 < PF \leq 0.90$ $\quad F_r^T \leq (3.50 - PF)((5.18 \times 10^{-7} \times FL) + 0.449)$

c. $PF \leq 0.70$ $\quad F_r^T \leq 1.75\ ((8.28 \times 10^{-7} \times FL) + 0.718)$

where:

PF = THERMAL POWER divided by RATED THERMAL POWER
FL = The lesser of either:

1) The reactor coolant flow rate measured per Specification 4.2.6.1 down to a minimum of 325,000 gpm, or

2) 340,000 gpm

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T exceeding its limit, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limit and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or
b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENT

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r\ (1+T_q)$ and F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
b. At least once per 31 days of accumulated operation in Mode 1, and
c. Within four hours if the AXIMUTHAL POWER TILT (T_q) is > 0.020.

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump Combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

*See Special Test Exception 3.10.2

AZIMUTHAL POWER TITL - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AXIMUTHAL POWER TILT (T_q) shall not exceed 0.02.

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

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be $\geq$ 0.02 but $\leq$ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the Total Integrated Radial Peaking Factor (F_r^T) is within the limit of Specification 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the Total Integrated Radial Peaking Factor (F_r^T) is within the limits of Specification 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to $\leq$ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 7 days when the Channel High Deviation Alarm is OPERABLE,

*See Special Test Exception 3.10.2.

DNB MARGIN

LIMITING CONDITION FOR OPERATION

3.2.6 The DNB margin shall be preserved by maintaining the cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX within the limits specified in Table 3.2-1 and Figure 3.2-4.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.6.1 The cold leg temperature, pressurizer pressure, and AXIAL SHAPE INDEX shall be determined to be within the limits of Table 3.2-1 and Figure 3.2-4 at least once per 12 hours. The reactor coolant flow rate shall be determined to be within the limit of Table 3.2-1 at least once per 31 days.

4.2.6.2 The provisions of Specification 4.0.4 are not applicable.

TABLE 3.2-1

DNB MARGIN

Parameter	LIMITS Four Reactor Coolant Pumps Operating
Cold Leg Temperature	≤ 549°F
Pressurizer Pressure	≥ 2225 psia*
Reactor Coolant Flow Rate	≥ 340,000 gpm**
AXIAL SHAPE INDEX	Figure 3.2-4

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

**Flow reductions to 325,000 gpm are compensated for by reductions in the F_r^T limit (Specification 3.2.3).

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	4	0	2	3, 4, 5	4
12. Underspeed - Reactor Coolant Pumps	4	2(a)	3	1, 2(e)	2

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(a) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 780 psia when all CEAs are fully inserted; bypass shall be automatically removed at or above 780 psia.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(d) Trip does not need to be operable if all the control rod drive mechanisms are de-energized or if the RCS boron concentration is greater than or equal to the refueling concentration of Specification 3.9.1.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) ΔT Power input to trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 5% of RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 4 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. ≤ 5% of RATED THERMAL POWER, immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

b. ≥ 5% of RATED THERMAL POWER, operation may continue with the inoperable channel in the bypassed condition, provided the following conditions are satisfied:

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Storage				
Criticality Monitor and Ventilation System Isolation	S	R	M	*
b. Control Room Isolation	S	R	M	ALL MODES
c. Containment High Range	S	R**	M	1, 2, 3, & 4
d. Noble Gas Effluent Monitor (high range) (Unit 2 Stack)	S	R	M	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Atmosphere-Particulate	S	R	M	ALL MODES
b. Containment Atmosphere-Gaseous	S	R	M	ALL MODES

*With fuel in storage building
**Calibration of the sensor with a radioactive source need only be performed on the lowest range. Higher ranges may be calibrated electronically.

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with at least one OPERABLE detector segment in each core quadrant on each of the four axial elevations containing incore detectors and as further specified below:

a. For monitoring the AXIMUTHAL POWER TILT:

At least two quadrant symmetric incore detector segment groups at each of the four axial elevations containing incore detectors in the outer 184 fuel assemblies with sufficient OPERABLE detector segments in these detector groups to compute at least two AZIMUTHAL POWER TILT values at each of the four axial elevations containing incore detectors.

b. For recalibration of the excore neutron flux detection system:

1. At least 75% of all detector segments,

2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

c. For monitoring the UNRODDED INTEGRATED RADIAL PEAKING FACTOR or the linear heat rate:

1. At least 75% of all incore detector locations,

2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

An OPERABLE incore detector segment shall consist of an OPERABLE rhodium detector constituting one of the segments in a fixed detector string.

An OPERABLE incore detection location shall consist of a string in which at least three of the four incore detector segments are OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

An OPERABLE quadrant symmetric incore detector segment group shall consist of a minimum of three OPERABLE rhodium incore detector segments in 90° symmetric fuel assemblies.

APPLICABILITY: When the incore detection system is used for:

a. Monitoring the AZIMUTHAL POWER TILT,

b. Recalibration of the excore neutron flux detection system, or

c. Monitoring the UNRODDED INTEGRATED RADIAL PEAKING FACTOR or the linear heat rate.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENT

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Monitoring the AZIMUTHAL POWER TILT.

2. Recalibration of the excore neutron flux detection system.

3. Monitoring the UNRODDED INTEGRATED RADIAL PEAKING FACTOR or the linear heat rate.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: ALL MODES.

ACTION:

a. With the number of OPERABLE seismic monitoring channels less than required by Table 3.3-7, restore the inoperable channel(s) to OPERABLE status within 30 days. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more seismic monitoring channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each of the above seismic monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, within 15 minutes initiate and continue boration at > 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at > 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT AND INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 are suspended, immediately:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1 or

b. Be in HOT STANDBY within 2 hours.

SURVEILLANCE REQUIREMENT

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 or 3.2.4 are suspended.

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 of 3.60% Δk/k is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 2% Δk/k shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 10,060 + 700/-0 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating 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 pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection flowpaths are provided to ensure single functional capability in the event an assumed failure of a pump or valve renders one of the flowpaths inoperable. Redundant flow paths from the Boric Acid Storage Tanks are achieved through Boric Acid Pumps, gravity feed lines and Charging Pumps. Redundant flow paths from the Refueling Water Storage Tank are achieved through Charging Pump flow path guaranteed by Technical Specification 3.1.2.2 and the HPSI flow path guaranteed by Technical Specification 3.5.2 and 3.5.3. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The minimum boration capability is sufficient to provide a SHUTDOWN MARGIN of 3.6% $\Delta k/k$ at all temperatures above 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires an equivalent of 4900 gallons of 3.5% boric acid solution from the boric acid tanks plus 15,000 of 1720 ppm borated water from the refueling water storage tank. The refueling water storage tank can also be used alone by feed-and-bleed using well under the 370,000 gallons of 1720 ppm borated water required.

The requirements for a minimum contained volume of 370,000 gallons of borated water in the refueling water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

3/4.1.2 BORATION SYSTEMS (Continued)

The analysis to determine the boration requirements assumed that the Reactor Coolant System is borated concurrently with cooldown. In the limiting situation when letdown is not available, the cooldown is assumed to be initiated within 26 hours and cooldown to 200°F is completed in the next 28 hours.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 2% Δ k/k SHUTDOWN MARGIN at 140°F after xenon decay. This condition requires either 3750 gallons of 2.5% boric acid solution from the boric acid tanks or 57,300 gallons of 1720 ppm borated water from the refueling water storage tank.

The maximum boron concentration requirement (3.5%) and the minimum temperature requirement (55°F) for the Boric Acid Storage Tank ensures that boron does not precipitate in the Boric Acid System. The daily surveillance requirement provides sufficient assurance that the temperature of the tank will be maintained higher than 55°F at all times.

A minimum boron concentration of 1720 ppm is required in the RWST at all times in order to satisfy safety analysis assumptions for boron dilution incidents and other transients using the RWST as a borated water source as well as the analysis assumption to determine the boration requirement to ensure adequate shutdown margin.

3/4.1.3 MOVEABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment (≥ 20 steps) of two or more CEAs, require a prompt shutdown of the reactor since either

3/4.1.3 MOVEABLE CONTROL ASSEMBLIES (Continued)

of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a immovable or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 20 steps) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (≥ 20 steps) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor protective system would not detect the degradation in the radial peaking factor and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with two OPERABLE excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2 using the Power Ratio Recorder. The power dependent limits of the Power Ratio Recorder are less than or equal to the limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.2, 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the Total Integrated Radial Peaking Factor does not exceed the limits of Specification 3.2.3.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.07, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02. Note the Items (1) and (4) above are only applicable to fuel batches "A" through "L".

3/4.2.3 and 3/4.2.4 TOTAL INTEGRATED RADIAL PEAKING FACTORS-F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_r^T and T_q are provided to 1) ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits, and, 2) ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS

setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F_r^T = F_r (1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_r^T and T_q are within their limits provide assurance that the actual values of F_r^T and T_q do not exceed the assumed values. Verifying F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.6 DNB MARGIN

The limitations provided in this specification ensure that the assumed margins to DNB are maintained. The limiting values of the parameters in this specification are those assumed as the initial conditions in the accident and transient analyses; therefore, operation must be maintained within the specified limits for the accident and transient analyses to remain valid.

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.17 during all normal operations and anticipated transients.

A single reactor coolant loop with its steam generator filled above 10% of the span provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

In MODES 4 and 5, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq$ 275°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 43°F (31°F when measured by a surface contact instrument) above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 296,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

3/4.4.3 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.4 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the reactor coolant system during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which occurs if the heaters are energized uncovered. The maximum water level in the pressurizer ensures that this paramter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish and maintain natural circulation.

The requirement that 130 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 54 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq$ 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50, with the option of the use of the mass point method for performing leakage calculations.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and 3.6.1.2. The limitations on the air locks allow entry and exit into and out of the containment during operation and ensure through the surveillance testing that air lock leakage will not become excessive through continuous usage.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that the containment peak pressure does not exceed the design pressure of 54 psig during LOCA conditions.

The maximum peak pressure obtained from a LOCA event is 53.8 psig. The limit of 2.1 psig for initial positive containment pressure will limit the total pressure to less than the design pressure and is consistent with the accident analyses.

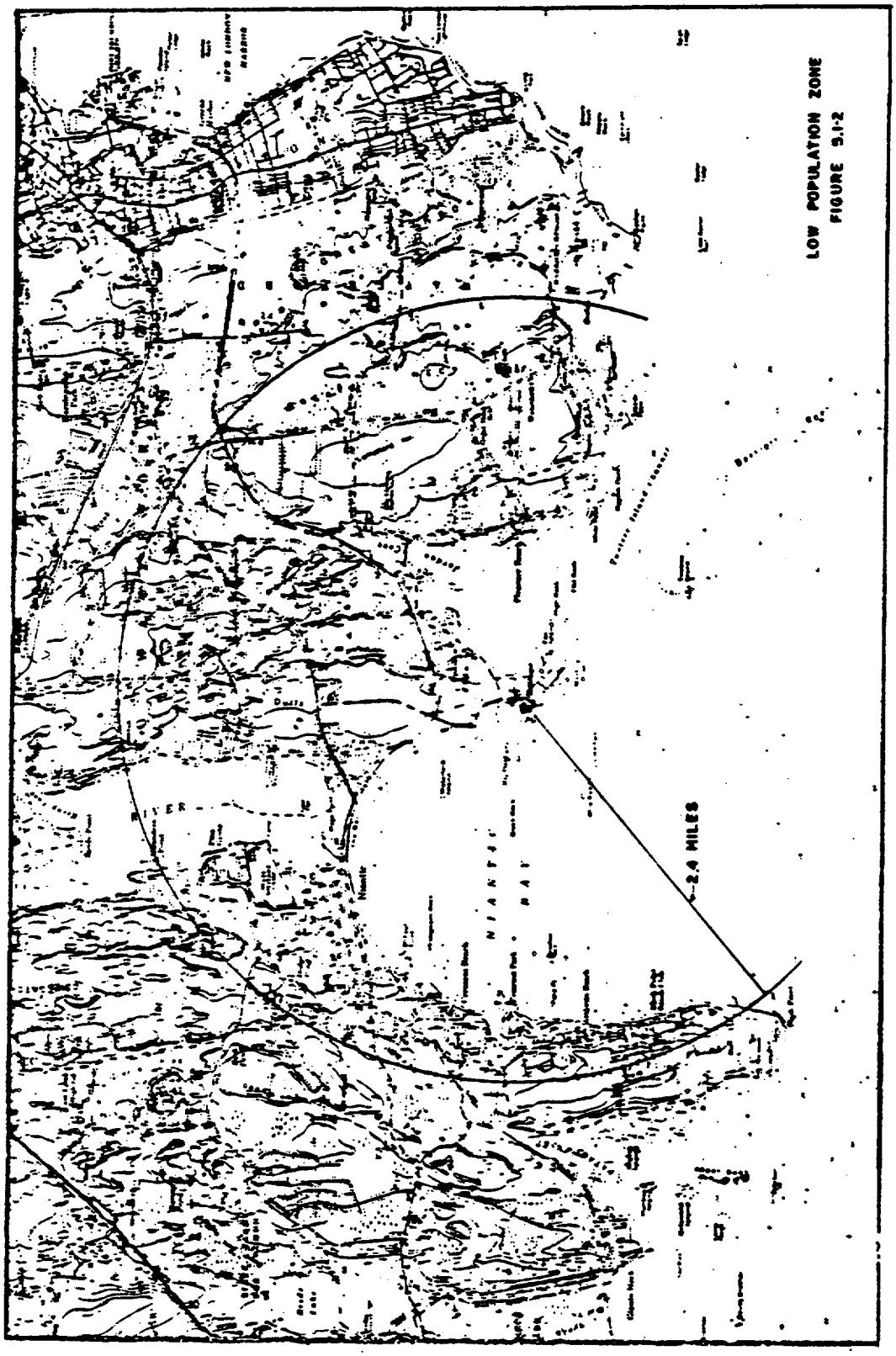
3/4.6.1.5 AIR TEMPERATURE

The limitation on containment air temperature ensures that the containment peak air temperature does not exceed the design temperature of 289°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 53.8 psig in the event of a LOCA. The measurement of containment tendon lift off force, the visual and metallurgical examination of tendons, anchorages and liner and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures."



LOW POPULATION ZONE
FIGURE 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 54 psig and a temperature of 289°F.

PENETRATIONS

5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.8 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing 176 rods. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent of U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and no part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the original design provisions contained in Section 3.0 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

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 4.2.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F except for the pressurizer which is 700°F.

ATTACHMENT TO LICENSE AMENDMENT NO. 139

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove	Insert
II	II
V	V
XI	XI
1-5	1-5
2-2	2-2
2-4	2-4
2-5	2-5
2-7	2-7
B2-1	B2-1
B2-3	B2-3
B2-5	B2-5
B2-6	B2-6
B2-7	B2-7
B2-8	B2-8
3/4 1-1	3/4 1-1
3/4 1-5	3/4 1-5
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 2-3a
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 2-7a	--
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 2-14	3/4 2-14
3/4 3-4	3/4 3-4
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 10-2	3/4 10-2
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 1-4	B 3/4 1-4
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
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BASES

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139

TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letters (Refs. 1 and 2) dated August 26, 1988 and November 15, 1988, Northeast Nuclear Energy Company (NNECO), the licensee for the Millstone Unit 2, submitted its safety analyses to support Cycle 10 operation. The reload application involves three fuel design related issues: (1) the replacement of 60 spent fuel assemblies with 60 fuel assemblies provided by Advanced Nuclear Fuels Corporation (ANF formerly Exxon Nuclear), (2) the analysis of safety considerations involved in the determination of Cycle 10 operating limits, and (3) incorporation of new limits on the linear heat generation rate (LHR). In support of the reload application, the licensee also provided a reload analysis submittal and supporting analyses. These submittals are as follows:

1. License amendment to Technical Specifications (Ref. 2).
2. Cycle 10 safety analysis report (Ref. 3).
3. Fuel Design Report (Ref. 4).
4. Transient analysis reports (Refs. 5 and 6).
5. Steam Line Break analysis report (Ref. 7).
6. Small break loss-of-coolant-accident (LOCA) analysis report (Ref. 8).
7. Large break LOCA analysis report (Ref. 9).

The proposed Technical Specification (TS) changes and the supporting analyses were based on an assumption that a minimum reactor coolant system (RCS) flow rate of 340,000 gpm, the current Technical Specification limit, can be maintained through the Cycle 10 operation. Subsequently, the licensee indicated (Ref. 29) that the RCS flow has been observed to be decreasing during the past several cycles of operation. As a result, the licensee stated that the margin between operating RCS flow rate and the existing TS limit is very small. In order to increase the margin in the RCS flow, the licensee submitted the request for the TS change to allow operation of the Cycle 10 core with a minimum RCS flow rate of 325,000 gpm. A supporting analysis (Ref. 30) ANF-89-011 (Millstone Unit 2, Reduced Flow, Standard Review Plan - Chapter 15 Event Analysis) was also submitted for review. The supporting analysis is to address the effect of the reduced RCS flow on the transient behaviors and the operating safety limits. In Reference 30, the licensee identified and analyzed the events affected by the reduced RCS.

The licensee's objective for these submittals is to demonstrate that the proposed Technical Specification (TS) changes are appropriately reflected in the assumptions used in the Cycle 10 design and the analyses of transients and LOCAs in order to support its position that the Millstone Unit 2 can be operated safely at a rated core thermal power of 2700 MWt throughout Cycle 10. The NRC staff has reviewed these submittals with analyses based on the current TS RCS flow rate of 340,000 gpm and the reduced RCS flow rate of 325,000 gpm for the reload application and prepared the evaluation as follows.

2.0 EVALUATION

2.1 Reload Description

Millstone Unit 2 core consists of 217 assemblies, each having a 14x14 fuel rod array. The assemblies are composed of up to 176 fuel rods, 4 control rod guide tubes and 1 center control rod guide tube/instrument tube. The Millstone 2 Cycle 10 (M2C10) core will retain 157 Westinghouse assemblies from the previous cycle and add 60 unirradiated ANF fuel assemblies referred to as Batch M, ANF-1. The average enrichment for the added rods containing no burnable absorbers is 3.30 W/o U-235 (3.00 W/o around the guide and instrument tubes and 3.45 W/o elsewhere). For the added rods containing 1.0 W/o gadolinia the enrichment is 2.85 W/o U-235. For rods containing 6.0 W/o gadolinia the enrichment is 2.10 W/o U-235. The total batch burnable absorber requirement for ANF fuel is 576 gadolinia bearing rods.

The fresh fuel is scatter-loaded throughout the core. The fresh assemblies loaded in the core interior contain gadolinia-bearing fuel in order to control power peaking and reduce the initial boron concentration to maintain the moderate temperature coefficient (MTC) within its Technical Specification limit. The exposed fuel is also scatter-loaded in the center in a manner to control the power peaking.

2.2 Fuel Design

The fresh ANF fuel assemblies are the first such ANF assemblies to be used in Millstone Unit 2. The design of ANF fuel assemblies are described in Reference 10, "Generic Mechanical Design Report - Exxon 14x14 Fuel Assemblies for Combustion Engineering Reactors," which was previously approved by the NRC. The licensee submitted, in Reference 4, the analysis pertinent to the Millstone Unit 2 ANF-1 reload fuel assemblies which are not covered in the generic report (Refs. 10 & 28). The analysis in Reference 4 is done using the NRC approved methods in References 11 and 12, "Qualification of Exxon Nuclear Fuel for Extended Burnup."

We have evaluated the analysis for fuel performance in Reference 4 and found that the fuel design parameters were calculated by using the NRC approved methods and were within expected ranges of a typical ANF fuel assembly in a PWR core. Therefore, we conclude the fuel design acceptable.

The proposed linear heat generation rate (LHGR) limit for the M2C10 reload is 15.1 Kw/ft. Since this approved limit is obtained by ANF using the NRC approved EXEM/PWR evaluation model (Ref. 13), is well below the design limit of 21 Kw/ft, and thus is acceptable.

2.3 Nuclear Design

The nuclear design for the Millstone Unit 2 Cycle 10 reload has been performed by ANF with the approved methodologies (Ref. 12), which include the MICBURN-2/CASMO-2E and XTGPWR codes. CASMO-2E is a two-dimensional transmission probability code for burnup calculations on PWR assemblies or pin cells. MICBURN-2 is a multi-group one dimensional transmission probability code which calculates the microscopic burnup in an absorber rod containing initially homogeneously distributed gadolinia and generates effective cross-sections as a function of the gadolinia number density to be used in CASMO-2E assembly depletion. The MICBURN-2/CASMO-2E code generates the cross-sections to the XTGPWR code (Ref. 14) which determines power and exposure distribution, reactivity feedback characteristics and cold shutdown margin. The results of the reload analyses are given in Reference 4. Since the M2C10 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The objective of this review is to confirm that the thermal-hydraulic design of the reload has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and transient conditions. The review includes two areas: (1) the core thermal-hydraulic design methodology, (2) thermal-hydraulic compatibility, and (3) minimum departure from nuclear boiling (DNB). The licensee has submitted a reload report in Reference 4 for the M2C10 operation. Discussion of the review of Reference 4 concerning thermal-hydraulic design is as follows.

2.4.1 Mixed Core Thermal-Hydraulic Design Methodology

The analytical tools used by ANF for the thermal-hydraulic design are the XCOBRA-IIIC code (Ref. 15 & 25) and the XNB critical heat flux (CHF) correlation (Ref. 16). Both methods were previously approved by NRC with a restriction that an adjustment of 2 percent of the minimum DNBR must be included for mixed cores containing hydraulically different fuel assemblies.

In the ANF thermal-hydraulic design analysis, the two steps in the calculations are firstly, an octant-core calculation is done on an assembly-by-assembly basis. In this analysis the limiting bundle is placed at its allowable maximum radial peak of 1.61 while the remaining assemblies are at 111.7 percent over power. Inlet flow maldistributions are accounted for by a reduction of 5 percent in the limiting assembly. Cross flow between adjacent assemblies in

the open lattice core is directly model. The assembly specific single-phase loss coefficients are used to hydraulically characterize the assemblies in the mixed core. The results of this calculation are the axial flow distribution for hot assembly and cross flow boundary conditions which will be used in the detailed subchannel model.

Next, an octant of the hot assembly is modeled on rod-by-rod basis to determine DNBR for the core. In this model cross flow between the limiting and adjacent fuel assembly is accounted for via cross flow between adjacent subchannels. As part of their subchannel analysis, ANF increases the peak rod heat flux by typically 3 percent to account for extremes in fuel rod manufacturing tolerances and uses a flat peaking distribution within the rod array except for the limiting rod which is placed at its maximum peak.

2.4.2 Hydraulic Compatibility of ANF and Co-resident Fuel

Hydraulic performance differences between ANF and Westinghouse (W) fuel were assessed with pressure drop test performed in ANF's hydraulic test facility. Using the loss coefficients from these tests, ANF determined that the overall assembly loss coefficient for the ANF assembly exceeds that of the Westinghouse fuel assembly by 21 percent at typical full power plant operation. For a mixed core, a larger hydraulic resistance causes a net flow diversion from the ANF assembly. This flow diversion from the ANF assemblies to the Westinghouse assemblies results in lower DNBRs for the ANF fuel and increased DNBRs for the Westinghouse fuel.

The staff finds that in accordance with our approved procedure, with an adjustment of 2 percent of the minimum DNBR to compensate for uncertainty in the mixed core methodology, the effect of the hydraulic differences between the ANF assemblies and Westinghouse assemblies on the calculated minimum DNBR are appropriately considered and are acceptable.

2.4.3 DNBR Safety Limit

The safety analyses for the Millstone Unit 2 Cycle 10 reload were done to analytically demonstrate that DNB can be avoided for the limiting rod in the core with 95% probability at 95% confidence level throughout each analyzed transient. The 95/95 DNBR safety limit is 1.17 for the XNB correlation after rod bow penalties and the 2 percent adjustment for uncertainties in mixed core methodology are applied. This DNBR safety limit was previously approved by the NRC staff, and therefore it is adequate for use in the M2C10 reload application.

2.5 Transient Analysis of FSAR Chapter 15 Events

The transient and accident analyses discussed in Sections 2.5 and 2.6 are based on assumed RCS flow of 340,000 gpm. The safety analysis based on the reduced RCS flow of 325,000 gpm is discussed in Section 2.7. Plant transient analysis

is included in References 5, 6, and 7 to support operation of M2C10 with a mixed core of 60 ANF fuel assemblies and 157 W fuel assemblies. Several physics parameters are more limiting than Cycle 9 due to the extension of the cycle length for ANF fuel to 18 months. These parameters include (1) increased shutdown margin 2.9 to 3.6% delta-k/k, (2) increased maximum radial peaking factor (1.537 to 1.61) at full power, and (3) increased both positive and negative bounds on the moderate temperature coefficient. In addition, the analysis also includes an effect of an anticipated need for a reduced core inlet temperature (12°F). The licensee evaluated (Ref. 3) all events described in Chapter 15 of the Standard Review Plan (SRP) and reanalyzed the events which are either limiting events or events with initiator or controlling parameters changed from the analysis of record so that the events need to be reanalyzed for current licensing application. The staff evaluation of the results of transient analysis (Refs. 6 & 7) is discussed as follows.

2.5.1 Increase in Steam Flow Event

This event is initiated by a failure of the main steam system that results in an increase in steam flow from the steam generator. To evaluate this transient, the licensee analyzed two cases: one at full power and one at the hot shutdown condition. In the analysis, the end-of-cycle reactivity feedback coefficients were used to maximize the challenge to the specific fuel design limits for both cases. Since the analytical results show that DNB is not expected to occur and the fuel centerline melt limit of 21 Kw/ft is not violated, the staff concludes the results are acceptable.

2.5.2 Loss of External Load Event

ANF analyzed this event using the approved PTS-PWR code (Ref. 17). Two cases were analyzed for this event: one maximizing the overpressurization, and one minimizing the fuel design limits. In both cases the input parameters were conservatively assumed to maximize the increase in reactor power during the transient. However, for the overpressurization event the parameters and safety system actuation set-points were assumed to maximize the system overpressure, while for the low DNBR event the parameters and safety system actuation set-points were assumed to minimize the pressurization in order to result in a minimum DNBR during transient. Since the results, using the approved code, show the peak pressure of 2697 psia, less than a limit of 2750 psia, a minimum DNBR of 1.39 and peak LHR of 17.6 Kw/ft resulting in no violation of the specific fuel design limits, the staff concludes that the results are acceptable.

2.5.3 Closure of a Single Main Steam Isolation Valve (MSIV)

From Reference 5, the limiting case is the event initiated from full power conditions. For simultaneous closure of both MSIVs, the event is similar to the event discussed in Section 2.5.2. The turbine stop valve closure time used in the Section 2.5.2 analysis (0.1 second) is much smaller than the MSIV closure time (6 seconds). Thus the consequences of event discussed in Section 2.5.2 will bound those of the dual closure event.

The asymmetric conditions resulting from the closure of only one of the two MSIVs is similar to a steam line break event since the primary coolant associated with the closed MSIV experiences a heatup due to loss of heat sink and the primary coolant loop associated with the open MSIV experiences a cooldown due to the load increase. The approved steam line break (SLB) methodology (Refs. 18 & 19) was used to perform this analysis. The neutronic parameters required to predict radial power distribution between the cold and the hot side of the core were based on the event specific XTGPWR (Ref. 14) calculations. These calculations differ from the SLB calculations due to the difference in the power range of interest. The end-of-cycle reactivity feedback coefficients were used to maximize the worst cooldown effect. The results indicated that the acceptance criteria are met. Since the minimum DNB limit is not exceeded by this event, the peak LHR is less than the 21 Kw/ft limit to centerline melt and the maximum pressure is less than 110% of design pressure, the staff concludes that the results are acceptable.

2.5.4 Loss of Feedwater Flow Event

Two cases were analyzed for this event: one maximizing pressurizer liquid level and one minimizing steam generator liquid inventory. The analysis was performed with the approved SLOTRAX-ML code (Ref. 20). The results showed that this event does not result in the violation of safety DNBR limit, peak pressurizer pressure does not exceed 110% of the design pressure and primary liquid is not discharged through the safety valves. It was also shown that the auxiliary feedwater system supplies adequate cooling water to allow a safe plant shutdown and prevent steam generator dryout. Based on the above analytical results, the staff concludes the analysis of the event is acceptable.

2.5.5 Loss of Forced Reactor Coolant Flow Event

This event is initiated by a loss of the power supplied to, or a mechanical failure of an RCS pump. As a result, the core flow rate will decrease and core temperature will increase. Prior to reactor trip, the combination of decreased flow and increased temperature may violate the DNBR safety limit. In the analysis, the assumptions were made to minimize pressure which minimizes DNBR. The steam bypass and atmospheric dump valves were both assumed not to operate to minimize the calculated DNBR. ANF used the approved statistical set-point methodology (Ref. 21) to evaluate the DNBR consequences of this event. The results (Refs. 6 & 22) showed that no DNBR safety limit is violated and that the maximum LHR of 17.2 Kw/ft is below the acceptable limit of 21 Kw/ft. The staff finds the results acceptable.

2.5.6 Reactor Coolant Pump Rotor Seizure Event

This event assumed the locked pump loss coefficient given by the homologous pump curve at zero pump speed. A statistical application of uncertainties demonstrated that the minimum DNBR for the loss of flow event is greater than XNB DNB limit. Due to the presence of margin to the DNB LCO limits for a loss

of RCS flow event and the inherent similarity between locked rotor and loss of RCS flow event, ANF concluded that fuel failures are precluded for the locked rotor event. The calculated peak LHR of 17.4 Kw/ft is less than the 21 Kw/ft limit to centerline melt. Since results demonstrated that no fuel failures were expected for this event, the staff concludes that the results are acceptable.

2.5.7 Control Rod Withdrawal Event

The PTS-PWR code (Ref. 17) was used to analyze an uncontrolled rod withdrawal for a reactivity insertion rate of $4x10^{-6}$ AK/K/sec from full power initial conditions. The minimum DNBR is 1.21 which is above the 95/95 acceptance limit of 1.17 using XNB DNB correlation. This transient tripped the reactor on thermal margin/low pressure signal. The maximum peak pellet LHR is calculated to be 19.1 Kw/ft. Since the code used for analysis was approved and the results showed sufficient margin existed to prevent fuel damage from a control rod withdrawal event, the staff concludes that the results are acceptable.

2.5.8 Control Rod Drop Event

In this event, the core power initially decreases due to the insertion of negative reactivity resulting from the dropped control rod. Moderator and Doppler temperature feedback cause power to return to its initial state. The event results in a localized increase in the radial peaking factor, which causes DNBR to decrease for the case with the initial core condition at full power. The DNBR consequences for this event were evaluated using the approved statistical set-point methodology (Ref. 21). The results showed that the DNBR safety limit will not be violated. Since the power initially decreases following the CEA rod drop event, no reactor trip occurs and protection of thermal margin limits is provided by the LCOs. The staff finds that the calculations using the approved methods show that fuel damage will not occur from a control rod drop event, therefore we conclude that the results are acceptable.

2.5.9 Single Control Rod Withdrawal Event

This event results in a reactivity insertion and a localized increase in the radial peaking factor. The degradation of core condition characteristics of reactivity insertion transient, combined with an increase in local radial peaking, poses a challenge to the DNBR limit. Since the calculational results show that the amount of fuel failure for this event (Ref. 23) is bounded by that of a control rod ejection event and the peak LHR is less than the acceptable limits of 21 Kw/ft, the staff concludes that the analysis is acceptable.

2.5.10 Boron Dilution Event

The boron dilution analysis was used to confirm that the required shutdown margins, which would enable the operator to have at least 15 minutes from the time of the first safety alarm until criticality for Modes 1 through 5, and 30 minutes for Mode 6 (the refueling mode), are as required in the SRP. These new shutdown margins were then reflected in the Technical Specifications. The calculated shutdown margin requirements to meet the required operator response time in the SRP are less than or equal to 3.6% delta-k/k for Modes 1 to 4, 2% delta-k/k for Mode 5 and 5% delta-k/k for Mode 6. Conservative assumptions were used in the analysis (i.e., the operator response time, to terminate the source of boron dilution flow, of 141 minutes instead of 15 minutes was used to determine the shutdown margin requirement for Mode 3 and the reactivity insertion rates at the lower bounded values were used to terminate power excursion for Modes 1 and 2). Therefore, the staff concludes the calculated required shutdown margins are adequate and acceptable.

2.5.11 Control Rod Ejection Event

The control rod ejection event was analyzed with the approved methods described in Reference 24. Energy deposition in the hot fuel pellet was evaluated for beginning of cycle and end of cycle conditions from hot zero power (HZP) and hot full power (HFP) initial conditions. In the analysis, no credit was taken for the flux flattening effects of reactivity feedback to maximize the total peaking factors. The results of analysis show that the HFP case results in a highest energy deposition of 240.6 cal/gm, which is below the acceptable limit of 280 cal/gm. An analysis of the core pressure surge associated with the control rod ejection indicates a maximum pressure of 2671 psia, below the acceptable limit of 2750 psia. The DNBR calculation shows that less than 11.5% of the core will experience fuel failure and the radiological consequences are within 10% of the 10 CFR 100 limits (Ref. 23). The staff finds that the approved methods were used to demonstrate that the primary system integrity will be maintained, the energy deposition is within the acceptable limit and the radiological release is within the 10 CFR 100 limits. Therefore, the NRC staff concludes that the results are acceptable.

2.5.12 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve Event

This event is primarily considered a depressurization event. The licensee determined that the limiting case is the event with the core operated at full power conditions, which was analyzed using the approved PTS-PWR code (Ref. 17) for system response, the XCOBRA-IIIC (Refs. 15 & 25) for the hot channel thermal hydraulic analysis and XNB correlation for determination of DNBR. The results demonstrate that the minimum DNBR is greater than the XNB DNB limit and the fuel centerline melt limit of 21 Kw/ft is not exceeded. Since the approved methods were used to show that the results are within the acceptable fuel design limits, the NRC staff concludes that the analysis is acceptable.

2.5.13 Steam Line Break Analysis

The licensee provided the results of the steam line break (SLB) analysis in Reference 9 for NRC staff review. The analysis was performed with the ANF SLB methods (Ref. 7), which were previously approved by the NRC staff (Ref. 19). The SLB methods used ANF-RELAP developed from RELAP5/MOD2 with ANF modifications, (Ref. 7) for system response, XTG (Ref. 14) for calculation of the core and hot assembly power distribution, XCOBRA-IIIC (Refs. 15 & 25) for determination of the core and hot assembly flow and enthalpy distributions, and the modified Barnett correlation (Ref. 26) for the DNBR calculations.

The licensee performed four double-ended guillotine SLB analyses in order to determine the limiting case for the consequences approaching the fuel design limits. The four cases were:

(1) A large SLB during full power operation in combination with a single failure, loss of offsite power and stuck CEA.

(2) Case 1 with offsite power available.

(3) A large SLB during HZP operation in combination with a single failure, loss of offsite power and stuck CEA.

(4) Case 3 with offsite power available.

The analyses determined that the HZP case with loss of offsite power is the limiting DNBR case, and the HZP with offsite power is the limiting case from standpoint of centerline melt. The worst calculated DNBR of 1.18 and peak LHR of 20.9 Kw/ft provide margin to fuel failure during SLB. The NRC staff concludes that the results are acceptable.

2.6 Loss-of-Coolant Accident Analysis

2.6.1 Small Break Loss-of-Coolant Accident (SBLOCA)

The licensee provided the results of the SBLOCA analysis in Reference 8. The SBLOCA analysis was performed with the NRC approved method, EXEM PWR Small Break Model (Ref. 27). The analysis was done assuming 102% of the core power of 2700 MWt, a maximum LHR of 15.1 Kw/ft and a radial peaking factor of 1.61. The licensee also assumed an average steam generator tube plugging of 23.5% and a maximum asymmetry of 5.9% in the analysis. Various break sizes (1%, 1.9%, 3% and 4% of double ended cold leg guillotine (DECLG) breaks) were performed and the results show that the limiting case is 1.9% of DECLG break with 12°F reduction in primary coolant temperature. The limiting case results in the highest peak cladding temperature of 1811°F, well below the acceptance criteria of 2200°F. The staff concludes that the SBLOCA analysis is acceptable since the approved method was used to show the analytical results to be within the acceptance criteria in 10 CFR 50.46.

2.6.2 Large Break Loss-of-Coolant Accident (LBLOCA) Analysis

The licensee provided the results of a large break LOCA analysis (Ref. 9). The LBLOCA analysis was performed with 102% of the core power of 2700 MWt, and an average steam generator tube plugging of 23.5% with a maximum asymmetry of 5.9%. In addition, the licensee assumed a primary coolant average temperature reduction of 12°F. Various break sizes (0.4, 0.6, 0.8 and 1.0 DECLG and double-ended-cold-leg-split (DECLS) were performed and results show that the worst break case is 0.6 DECLG break, resulting in a peak cladding temperature (PCT) of 2163°F of 2176°F for cases without and with 12°F reduction in primary coolant temperature, respectively. The analysis was performed with the NRC approved ANF EXEM/PWR evaluation model (Ref. 13). The evaluation model used RODEX2 for computation of initial fuel stored energy, fission gas release, and gap conductance; RELAP4-EM for the system and hot channel blowdown calculations; CONTEMPT/LT-22 for computation of containment back pressure; REFLEX for computation of system reflood and TOODEE2 for the calculation of fuel rod heatup during the refill and reflood portions of the LOCA transient.

The staff has reviewed the analysis. As a result, the staff found that the approved analytical methods and computer codes were used and the results show that the peak cladding temperature, metal-water reaction and clad oxidation are within the acceptance criteria in 10 CFR 50.46 for LOCA analysis. The NRC staff, therefore, concludes that the results of the LBLOCA analysis are acceptable.

2.7 Effects of RCS Flow Reduction on Safety Analyses

In order to increase the margin in the RCS flow the licensee proposed TS changes based on an assumed RCS flow of 325,000 gpm. In the supporting analysis (Ref. 30), the licensee identified and reanalyzed five events which are significantly affected by the reduced RCS flow. These events are: (1) loss of flow, (2) locked rotor, (3) control rod, (4) small break LOCA, and (5) large break LOCA.

The loss of flow event is reanalyzed because it is the limiting DNBR transient and demonstration of margin to the DNBR safety limit for this event will demonstrate sufficient margin for the other events. The locked rotor event is reanalyzed because in the original analysis, fuel failures were precluded for this event based on the available margin to the DNBR safety limit calculated in the loss of flow event. The reduction in the RCS flow has removed that margin. Thus, analysis of the potential fuel failure is needed for the locked rotor event. The control rod ejection event is reanalyzed because it is the limiting event with respect to the amount of predicted fuel failures. Reanalysis allows an assessment of the effect of the reduced RCS flow on the magnitude of the predicted fuel failures. The small and large break LOCAs are reanalyzed because an increase in the predicted PCTs are expected as a result of the decrease in RCS flow rate.

In the analysis, the following assumptions were made in order to compensate for the effect of the reduced RCS flow on the DNBR, PCTs and the predicted fuel failures:

1. Reduce the radial peak factor (Fr^T) from 1.61 to that specified in Table 15.0.5-1 of Reference 30.

2. Raise the thermal margin/low pressure trip set-point from 1750 psia to 1850 psia.

3. Alter the low power density LCO to allow power operation only to an axial shape index (ASI) of -0.04 instead of -0.06.

4. Reduce the LHR limit of 15.1 Kw/ft to 14.5 Kw/ft for flow rates greater than or equal to 325,000 but less than 340,000 gpm.

3.0 TECHNICAL SPECIFICATION CHANGES

Various changes to the Technical Specifications have been proposed in order to operate the M2C10 core. The changes include changes in (1) linear heat rate, (2) total integrated radial peaking factor, (3) total planar radial peaking factor, (4) moderator temperature coefficient, (5) shutdown margin and (6) low steam generator trip set-point. These changes include only the TS changes relating to the application dated November 15, 1989. Changes related to the reduced RCP flow rate as requested by letter dated February 1, 1989 will be in a subsequent amendment. Our assessment of the proposed Technical Specifications (TS) changes is summarized as follows.

1. Pages II, V and XI of the index - Deletes references to the unrodded planar radial peaking factor (Fxy) because Fxy is removed from the proposed Technical Specifications.

2. Page 1-5 of the definitions - Deletes the definition for Fxy because Fxy is being removed from the Technical Specifications.

3. Section 2.1.1 (Figure 2-2 on page 2-2) - A footnote is added to specify the minimum reactor vessel flow of 325,000 gpm for the case with the reduced Fr as specified in Technical Specification 3.2.3. This change is consistent with the assumptions used in the acceptable analytical results (Ref. 30) and is acceptable.

4. Section 2.2.1 - Increases the steam generator low pressure trip set-point and allowable value (Table 2.2-1, page 2-4) from 500 to 680 psia and 492 to 672 psia, respectively. The changes are to assure adequate protection against the asymmetric secondary side transient resulting from the enclosure of a single MSIV. Footnote 2 of Table 2.2-1 (page 2-5) is changed to allow the steam generator low pressure trip to be manually bypassed from 600 to 780 psia. This change is to assure that the reactor is in safe condition whenever the trip is bypassed. Supporting analysis

(Ref. 6) was performed to demonstrate the acceptability of the changes. The changes are acceptable. [A footnote to specify the design flow rate which is a base for the low RCS flow trip set-point. When the measured flow rates are greater than 340,000 gpm, 340,000 gpm is used as the design flow. When the measured flow rate is between 325,000 and 340,000 gpm, the measured flow rate is used as the design flow.] The thermal margin/low pressure trip set-point is raised from 1750 to 1850 psia. The changes are supported by the acceptable results (Ref. 30) and are acceptable.

5. Five changes to the bases of the Safety Limits and Limited Safety System Settings are as follows:

   (i) Page B 2-1 - Changes the references for the DNB correlation to be consistent with the XNB correlation used by ANF instead of the W-3 correlation used by Westinghouse.

   (ii) Pages B 2-1, 3, 5, 6 and 8 - changes the DNBR limit determined by using statistical methods from 1.30 to 1.17 to be consistent with the DNBR safety limit of the XNB correlation used by ANF.

   (iii) B 2-5 - Deletes the steam generator operating pressure of 815 psia. A qualitative statement without specific number provides clear definition of the technical base and is acceptable.

   (iv) Page B 2-7 - Changes the uncertainty factor for the thermal margin/low pressure trip from 67 to 72 psi which consists of a 22 psi pressure measurement error and a 50 psi time delay allowance. The change is consistent with the assumption used in the transient analysis.

   (v) Page B 2-5 - Changes the steam generator low pressure set-point from 500 psia to 680 psia to be consistent with the assumptions used in the supporting analysis for the reload application.

6. Section 3/4.1.1.1 (page 3/4 1-1) - Changes the shutdown margin for Modes 1 through 4 from 2.90% to 3.60% delta-k/k to reflect the Cycle 10 fuel characteristics. The change is supported by the transient analytical results and is acceptable.

7. Section 3.1.1.4 (page 3/4 1-5) - The most positive moderator temperature coefficient (MTC) for the power less than or equal to 70% of the full power and the most negative MTC at full power are changed from 0.5×10^{-4} to 0.7×10^{-4} delta-k/k/°F, and from -2.4×10^{-4} to -2.8×10^{-4} delta-k/k/°F, respectively. The changes are to reflect the Cycle 10 fuel characteristics and are acceptable.

8. Section 3/4.2.2 (pages 3/4 2-5 through 3/4 2-8) - Deletes the entire section regarding total planar radial peaking factor (Fxy) from the Technical Specifications. This deletion is acceptable because ANF's 3-D

power distribution methodology (Ref. 31) does not require this parameter (Fxy). The axial shape index (ASI) tents in this Section are required for monitoring of the LHR. The LHR tent is therefore moved to the LHR specification (Section 3/4.2.1).

9. Section 3/4.2.1 - The maximum linear heat rate (MLHR) is changed from 15.6 Kw/ft to 15.1 Kw/ft for the RCS flow rates greater than 340,000 gpm. The MLHR is 14.5 Kw/ft for the RCS flow rates greater than or equal to 325,000 gpm and less than 340,000 gpm. Also, the end of cycle coastdown restrictions are removed. The changes are supported by ANF in their LOCA and set-point analyses and are acceptable (Refs. 6 & 30). Figure 3.2-1 on page 3/4 2-3 with the current LHR value is deleted and the constant value is written into the text on pages 3/4 2-1 and 3/4 2-2. The definition of LHR on Figure 3.2-1 is also written into the text. Inclusion of the LHR limit in the text and deletion of the figure is proposed for simplification and does not affect the interpretation of the Technical Specification and is acceptable.

The LHR and the ASI requirements for monitoring of the LHR on Figure 3.2-2a are retained. Figure 3.2-2a is renumbered as 3.2-2 (Ref. 29). Figure 3.2-2b is deleted from the Technical Specification. The changes are consistent with the assumptions used for LOCA (Refs. 9 & 30) and set-point analyses and are acceptable.

The specific changes to the Technical Specifications for LHR monitoring using the excore detectors are discussed as follows:

(i) Item a of Section 3.2.1 is deleted. The deletion of item a, specifying the maximum allowable power less than 100% of rated thermal power at certain Fxy values, is consistent with the proposed Technical Specifications with deletion of Fxy.

(ii) Item b of Section 3.2.1 is combined with the introductory sentence. References to the maximum allowable power limit are deleted as discussed in item (i) above. The two separate conditions following the "either" are separated as Items a and b. References to Technical Specification 3.2.2 are changed to refer to Figure 3.2-2 as discussed above.

(iii) Section 4.2.1.2.b (page 3/4 2-2) - Reference to Technical Specification 3.2-2 is changed to Figure 3.2-2 as discussed above.

(iv) Section 4.2.1.2 (page 3/4 2-2) - Deletes Item C which requires verification every 31 days of operation within the limits of Figure 3.2-2. Since the limits of figure of 3.2-2 are monitored continuously by the power ratio recorder per Technical Specification 3.2.1, we find that Item C is redundant to a more restrictive requirement and deletion of Item C is acceptable.

10. Section 4.2.1.3 (page 3/4 2-2) - Deletes two penalty factors. They are (1) Item 4.2.1.3.b.1, the flux peaking augmentation factor as shown in Figure 4.2-1 and (2) Item 4.2.1.3.b.4, the LHR uncertainty factor of 1.01 due to axial fuel densification and thermal expansion. Based on the following reasons, we find the deletions acceptable.

    (i) The flux peaking augmentation factors are a result of the fuel pellets interacting with the cladding prior to the time of maximum duel densification. Since the licensee stated that the pellet and the clad will not occur prior to the time of maximum pellet densification in the ANF fuel and the NRC has previously approved the removal of this penalty for the ANF fuel used in the Combustion Engineering reactor, we conclude that the deletion of the flux peaking augmentation factors is acceptable.

    (ii) The licensee stated that the LHR uncertainty factor due to axial densification and thermal expansion is included in the engineering uncertainty factor of 1.03 (Item 4.2.1.3.6.3 on page 3/4 2-2) in the ANF methodology (Ref. 4). We, therefore, agree that this penalty does not need to be included again in the Technical Specification for the ANF fuel.

    The removal of these penalties only applies to the ANF fuel. They should still be applied to the Westinghouse and Combustion Engineering fuel when used in the core. Therefore, footnote is applied to these two factors (Item 4.2.1.3.b.1 and 4.2.1.3.b.4 on page 3/4 2-2) specifying that they only apply to the non-ANF fuel. This footnote is also added to Figure 4.2-1 (page 3/4 2-4).

11. Section 3.2.3 (page 3/4 2-9). The Total Integrated Radial Peaking Factor (Fr) of 1.537 is changed to 1.61. Figure 3.2-2.b is replaced by Figure 3.2.2. These changes are acceptable.

12. Section 3.2.4 (page 3/4 2-10) - Deletes the references to Fxy and the current Technical Specification Section 3.2.2 to be consistent with Item 8 of the proposed Technical Specifications discussed above. In this page, two references to "Total Integrated Radial Peaking Factor" are no longer all capitalized to be consistent with the term used in the Technical Specifications.

13. Section 3.3.1 - Changed footnote B on page 3/4 3-4 to be consistent with the changes regarding the set-point to bypass the steam generator low pressure trip as discussed in Item 3 of the proposed Technical Specifications.

14. Section 3.2.6 (page 3/4 2-14) - A footnote to the RCS flow of 340,000 gpm is added to indicate that the flow can be reduced to 325,000 gpm if there are reductions in the Fr as given in Technical Specification 3.2.3.

15. Section 3/4.3.3.2 (pages 3/4 3-30 and 3/4 3-31) - Deletes three references to the unrodded planar radial peaking factor (Fxy) to be consistent with the deletion of Fxy.

16. Section 3/4.10.2 (page 3/4 10-2) - Deletes test exceptions allowed from the current Technical Specification Section 3.2.2 (Fxy) to be consistent with the deletion of Fxy. Delete test exceptions allowed from Technical Specification 3.1.3.2 which dealt with the post length control rods because both the post length rods and Technical Specification were removed for Cycle 2.

17. Section 3/4.1 (pages B 3/4 1-1 and B 3/4 1-2) - The base section for shutdown margin is changed to 3.6% delta-k/k from its current value to be consistent with the proposed Technical Specification changes discussed in Item 6 above. Page B 3/4 1-4 refers to radial peaking factors. Since one of these two factors is deleted, the word "factor" is no longer pluralized.

18. Section 3/4.2 (page B 3/4 2-1) - Two changes are made in this section as follows:

    (i) When the LHR is monitored by the excore detectors, the allowable radial power distribution is currently given by the total planar radial peaking factor of Technical Specification 3.2.2. To be consistent with the proposed Technical Specifications, this is changed to the Total Integrated Radial Peaking Factor of Section 3.2.3.

    (ii) The end of cycle coastdown restrictions are removed. Therefore, the basis for this restriction is deleted. References to Fxy and Technical Specification 3/4.2.2 are deleted to be consistent with the deletion of Fxy and Technical Specification 3/4.2.2 discussed in Item 8.

19. Section 3/4.4.1 (page B 3/4 4-1) - Change the DNBR limit of 1.30 to 1.17 to be consistent with the proposed Technical Specification changes as discussed in Item 5(i).

20. Section 5.2-2 (Page B 3/4 6-2 and 5-4) - Correct the maximum design temperature in the containment building from 288°F to 289°F to be consistent with the value specified in the FSAR Section 6.4.1.1.

We have reviewed the Technical Specification changes and found that all of the changes reflect the characteristics of fuel in M2C10 and are supported by the assumptions used in the acceptable analytical results (Refs. 5 through 9 & 30) we therefore, conclude the Technical Specification changes acceptable.

4.0 SAFETY ANALYSES CONCLUSIONS

We have reviewed the reports submitted for the Cycle 10 reload of Millstone Unit 2 with the mixed core of the Westinghouse and ANF fuel assemblies, and the licensee's analytical results for transients and LOCAs. Based on this review,

we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload are supported by the acceptable analytical results. The operating limits associated with those changes and reload parameters are acceptable.

Table 1 summarizes a comparison of the consequences for the events analyzed to support the licensee's amendment request based on the RCS flow of 340,000 gpm (Ref. 1) versus consequence calculated for the reduced flow analysis. The NRC staff has reviewed the safety analysis (Ref. 30) with the reduced RCS flow of 325,000 gpm and found that the assumptions are consistent with the proposed TS changes and the approved methods were used to demonstrate that the applicable fuel performance acceptance criteria are met in all cases. Therefore, the staff concludes that the results in Reference 30 support the operation of the N2C10 core at a rated thermal power of 2700 MWt with average steam generator tube plugging up to 23.5% of the steam generator tubes.

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

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2. Letter with an attachment from R. J. Mrcozka (NNECO) to NRC, Millstone Nuclear Power Station, Unit No. 2 - Proposed License Amendment, Cycle 10 Reload, dated November 15, 1988.

3. ANF-88-126, Millstone Unit 2 Cycle 10 Safety Analysis Report, October 1988.

4. ANF-88-088 (Revision 1), Design Report for Millstone Unit 2 Reload ANF-1, August 1988.

5. ANF-87-161, Millstone Unit 2 Plant Transient Analysis Report - Analysis of Chapter 15 Events, September 1988.

6. ANF-87-161, Supplement 1, Millstone Unit 2 Plant Transient Analysis Report - Analysis of Chapter 15 Events, October 1988.

7. ANF-88-127, Millstone Unit 2 Steam Line Break Analysis, October 1988.

8. ANF-88-129, Millstone Unit 2 Small Break LOCA Analysis, October 1988.

9. ANF-88-118, Millstone Unit 2 Large Break LOCA/ECCS Analysis, August 1988.

10. XN-NF-82-09(P)(A), Generic Mechanical Design Report Exxon Nuclear 14x14 Fuel Assemblies for Combustion Engineering Reactors, dated November 18, 1983.

11. XN-NF-81-58(A), Rev. 2, RODEX2 - Fuel Rod Thermal-Mechanical Response Evaluation Model, dated March 1984.

12. XN-NF-82-06(A), Rev. 1, Supplements 2, 4, and 5, Qualification of Exxon Nuclear Fuel for Extended Burnup (PWR), dated October 1986.

13. Letter from D. Crutchfield (NRC) to G. Ward (ENC), Safety Evaluation of Exxon Nuclear Company's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Report, dated July 8, 1986.

14. XN-CC-28(A), Rev. 3, XTG: A Two Group Three Dimensional Reactor Simulator Utilizing Loose Mesh Spacing (PWR Version), January 1975.

15. XN-NF-82-21(P)(A), Rev. 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, September 1983.

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17. XN-NF-74-5(A), Rev. 2 and Supplements 3-6, Description of Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR), October 1986.

18. ANF-84-93 (Supplement 1), Steam Line Break Methodology for PWRs, June 1988.

19. Letter from A. Thadani (NRC) to R. Copeland (ANF), Acceptance for Referencing of Licensing Topical Reports, ANF-84-93(NP), and ANF-84-93(P), Supplement 1, Steam Line Break Methodology for PWRs, dated December 28, 1988.

20. XN-NF-85-24(A), SLOTRAX-ML: A Computer Code for Analysis of Slow Transients in PWRs, September 1986.

21. XN-NF-507(A), Supplements 1 and 2, ENC Setpoint Methodology for CE Reactors: Statistical Setpoint Methodology, September 1986.

22. Letter from E. Mroczka (NNECO) to NRC, Response to NRC Questions, dated February 3, 1989.

23. Letter from E. Mroczka (NNECO) to NRC, Response to NRC Questions, dated January 13, 1989.

24. XN-NF-78-44(A), A Generic Analysis of the Coolant Rod Ejection Transient for Pressurized Water Reactors, October 1983.

25. XN-NF-75-21(A), Rev. 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation, January 1986.

26. IN-1412(TID), A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia, July 1970.

27. Letter from A. Thadani (NRC) to G. Ward (ANF), Acceptance for Referencing Topical Report SN-NF-82-49(P), Revision 1, Exxon Nuclear Company Evaluation Model - EXEM/PWR Small Break Model, dated July 12, 1988.

28. Letters from E. Mroczka (NNECO) to NRC, Response to Additional Questions, dated January 13 and January 23, 1989 (proprietary and non-proprietary versions).

29. Letter with attachment from E. Mroczka (NNECO) to NRC, Proposed Revision to Techincal Specifications - Reduced Reactor Coolant System Flow Rate, dated February 1, 1989.

30. ANF-89-011, Millstone Reduced Flow, Standard Review Plan, Chapter 15 Event Analysis, January 1989.

31. ANF-507 (Addendum 1), Advanced Nuclear Fuels Corporation Setpoint Methodology for C.E. Reactors: Three-Dimensional Axial Power Distribution Generation, June 1988.

Principal Contributor: S. Sun

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