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NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

P.O. BOX 270
HARTFORD, CONNECTICUT 06101
(203) 666-6911

August 25, 1980

Docket No. 50-336
A01155

Director of Nuclear Reactor Regulation
Attn: Mr. Robert A. Clark, Chief
Operating Reactor Branch #3
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: (1) W. G. Counsil letter to R. Reid, dated March 6, 1980.
(2) T. M. Novak letter to W. G. Counsil, dated August 6, 1980.
(3) W. G. Counsil letter to R. A. Clark, dated August 7, 1980.
(4) W. G. Counsil letter to R. A. Clark, dated August 7, 1980.
(5) W. G. Counsil letter to R. Reid, dated February 12, 1980.
(6) W. G. Counsil letter to R. Reid, dated March 27, 1979.
(7) W. G. Counsil letter to R. A. Clark, dated June 2, 1980.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Additional Information on Cycle 4 Reload

In Reference (1), Northeast Nuclear Energy Company (NNECO) docketed the Basic Safety Report in support of Cycle 4 operation of Millstone Unit No. 2.

Reference (2) requested that NNECO provide the NRC Staff with additional information to complete the review of the thermal-hydraulics and transient and accident analyses sections of Reference (1). In addition, additional information was requested to complete the review of the reactor physics and fuels sections in the Reload Safety Analysis and the small and large break LOCA/ECCS performance results.

NNECO provided the response to Enclosure 1 of Reference (2) in References (3) and (4).

In response to Enclosure 2 of Reference (2), NNECO provides Attachment 1.

We trust you find this information satsifactory to resolve all questions received to date regarding Cycle 4 operation at Millstone Unit No. 2.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

Attachment

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Docket No. 50-336

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Additional Information on Cycle 4 Reload

August, 1980

1. Provide a list of physics tests to be performed during Cycle 4 testing, including the acceptance criteria for each test as well as the actions to be taken if the acceptance criteria are not met.

RESPONSE

In References (5) and (6), NNECO provided the Staff with a description of the start-up test program for Cycle 3. The Cycle 4 start-up test program will be identical to the program conducted for Cycle 3 with the exception of the power coefficient measurement.

Power coefficient measurement difficulties during the Cycle 3 start-up test program required that the test procedure be revised. The procedure is currently being revised for possible use during the Cycle 4 start-up test program however, the degree of readiness of the revised procedure will determine whether or not the power coefficient test is performed during Cycle 4 start-up testing.

The Power Coefficient test is not mandatory and NNECO has performed the test for informational purposes only.

Proposed changes to the Cycle 3 acceptance criteria for Cycle 4 are:

(a) The measured sum of all control banks should be equal to or greater than 90% of the predicted sum. The review criteria for individual CEA worth should be the greater of $\pm$ 15% or 100 pcm from the predicted value.

(b) For power distributions, the acceptance criteria of $\pm$ 10% on RPD should be changed to review criteria with the following values:

(1) $\pm$ 10% $(\frac{M-P}{P})$ for RPD > 0.9.

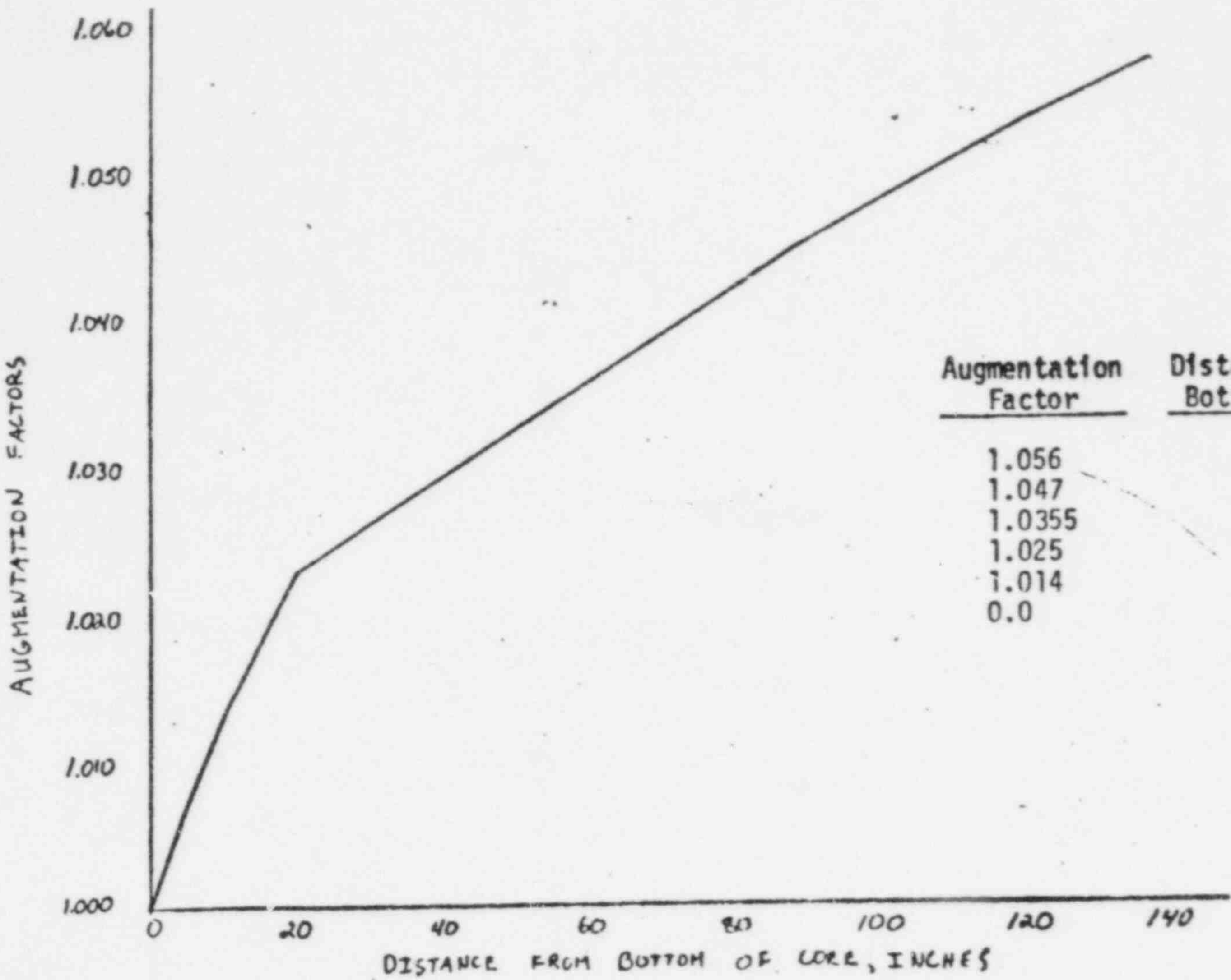
(2) $\pm$ 15% $(\frac{M-P}{P})$ for RPD $\leq$ 0.9.

(c) The equivalent reactivity difference between measured and predicted boron concentrations should be less than $\pm$ 1% $\Delta k/k$.

2. Previous cycles have used an augmentation factor to account for the power density spikes due to axial gaps caused by fuel densification. These previous cycle augmentation factors were included in the determination of F_{xy}. How are densification spikes accounted for in Cycle 4?

<u>RESPONSE</u>

Power peaking augmentation factors shown in attached Figure 4.2-1 will be used for Cycle 4. They were included in the determination of F_Q for all accident analyses performed for Cycle 4. The Technical Specification limits on local power density (Figure 2.2-2), LOCA peak linear heat rate (Figure 3.2-1), and LOCA allowable power level (Figure 3.2-2) also account for the augmentation factors.



Augmentation Factor	Distance from the Bottom of Core
1.056	136.6 in
1.047	100.0
1.0355	60.0
1.025	20.0
1.014	10.0
0.0	0.0

FIGURE 4.2-1 - AUGMENTATION FACTORS VS. DISTANCE FROM BOTTOM OF CORE

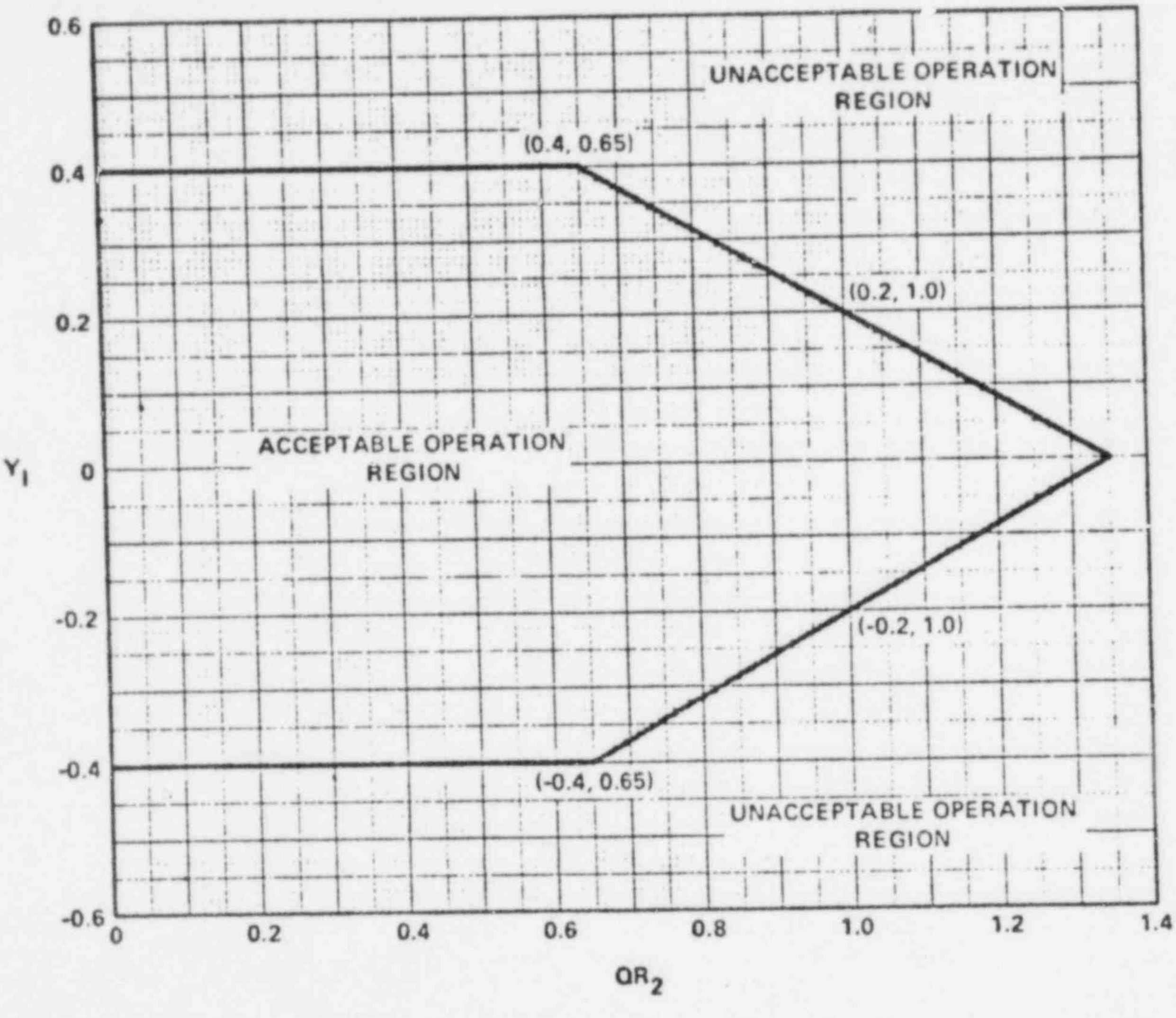


FIGURE 2.2-2 Local Power Density – High Trip Setpoint Part 2 (QR_2 Versus Y_I)

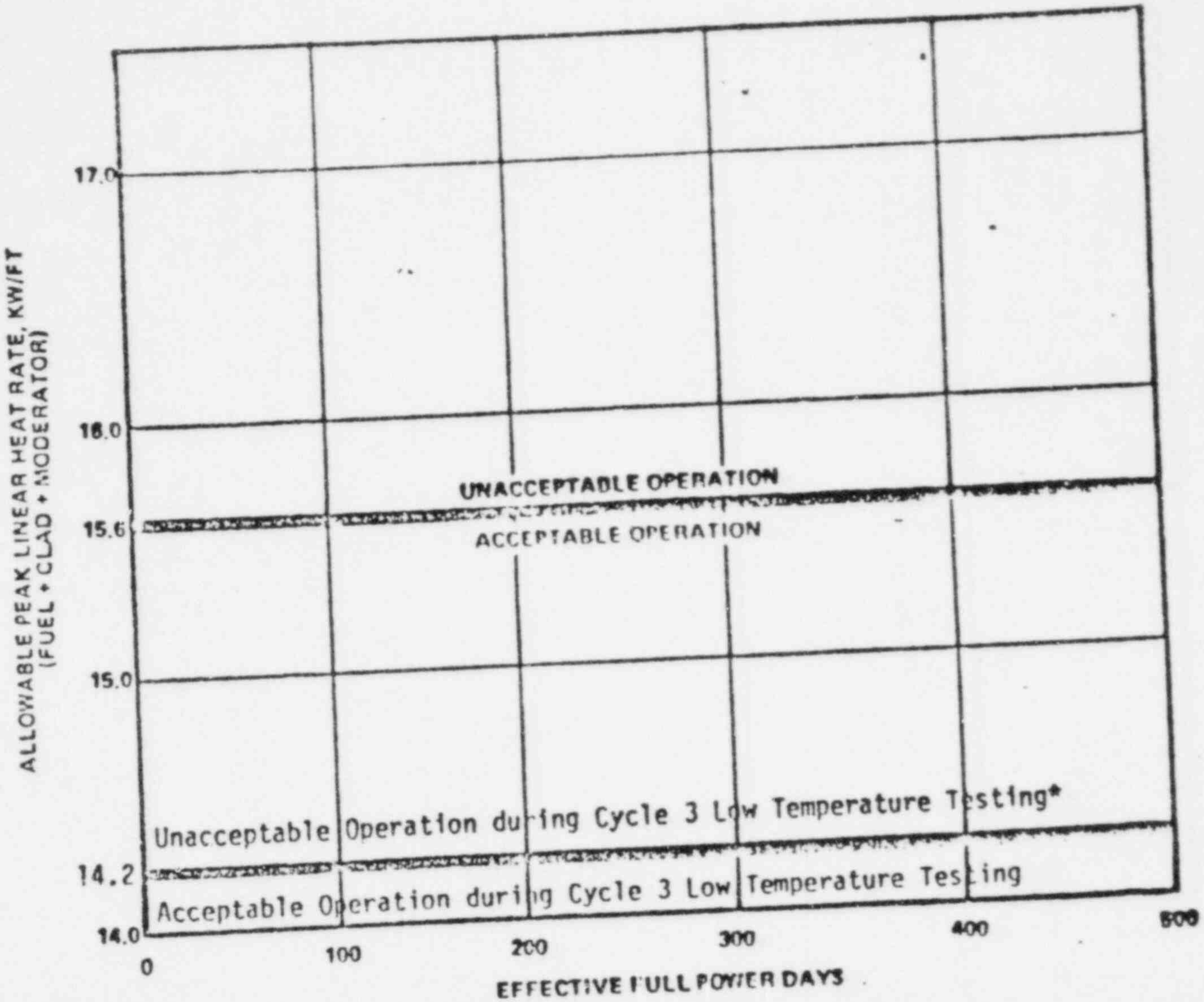


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Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

*During Cycle 3, low temperature testing is authorized for periods not exceeding 24 hours with the inlet temperature greater than or equal to 537oF and without varying the programmed pressurizer level. 43

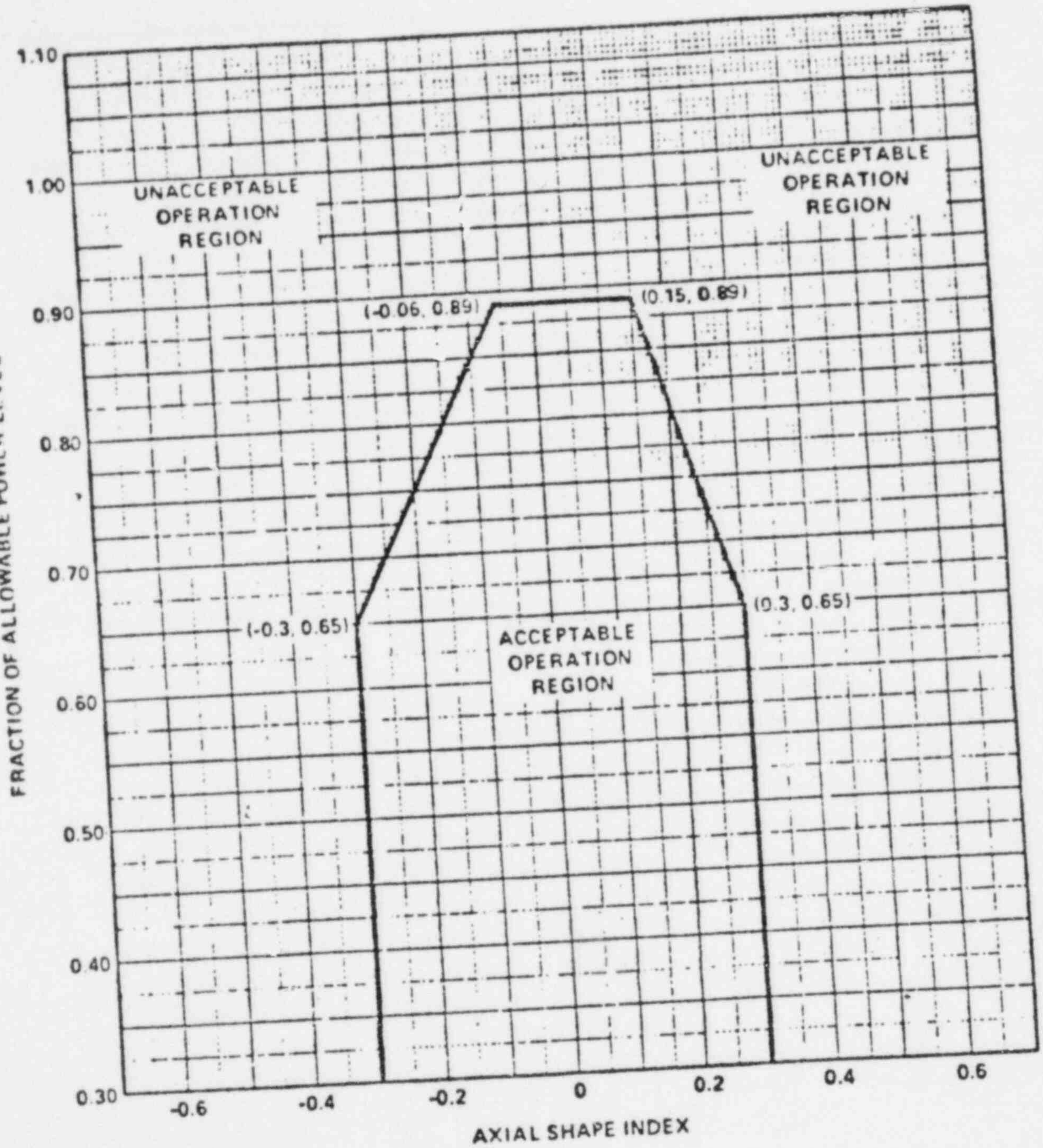

FIGURE 3.2-2 AXIAL SHAPE INDEX vs Fraction of Allowable Power Level per Specification 4.2.1.2c

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3. A partial list of physics characteristics for Cycles 2 and 3 and preliminary Cycle 4 data was presented in the BSR. Provide a list of final Cycle 4 physics characteristics and comparisons with previous cycle values including the maximum radial power peaks expected to occur (F_r and F_{xy} with uncertainties and biases).

RESPONSE

A comparison of Cycle 3 and final Cycle 4 physics characteristics is shown in Table 1. In Table 2 comparisons of F_r and F_{xy} with uncertainties and biases are given.

TABLE I

SUMMARY OF CORE PHYSICS CHARACTERISTICS

	Cycle 3 (1)	Cycle 4
Boron Concentration (ppm)		
HZP-BOL, No Xe, Peak Sm, ARO	1205	1339
HZP-BOL, No Xe, Peak Sm, Bank 7 In		1271
HFP-BOL, No Xe, Peak Sm, ARO		1248
HFP, Eq. Xe at 150 MWD/MTU, ARO	830	1000
HZP-BOL, ARI, $K \leq 0.99$		675
Refueling C_B, ARI, $K \leq 0.90$ [68°F]		$\geq$2000
Inverse Boron Worth (ppm/%Δρ)		
HZP-BOL	91 (2)	94
HFP-BOL	93	98
HFP-EOL	82	82
Control Rod Worths (-%Δρ)		
HZP-BOL, Bank 7 In	0.64	0.73
HZP-BOL, ARI		8.18
HFP-150 MWD/MTU, Bank 7 In	0.66	0.75
HZP-EOL, ARI (w/HFP Eq. Xe)		9.08
Moderator Temperature Coefficient (pcm/°F)		
HZP-BOL, ARO	5.4 (2)	3.8
HFP-BOL, ARO, Eq. Xe	-2.0	-4.2
HFP-EOL, ARO	-18.0	-23.6
Doppler Coefficient (pcm/°F)		
HZP-BOL, ARO	-1.44	-1.80
HFP-BOL, ARO	-1.13	-1.20
HFP-EOL, ARO	-1.22	-1.31

TABLE I (Con't.)

SUMMARY OF CORE PHYSICS CHARACTERISTICS

	Cycle 3 (1)	Cycle 4
Total Delayed Neutron Fraction, $\bar{\beta}_{eff}$		
HZP-BOL, ARO	0.00624	0.00584
HZP-EOL, ARO	0.00524	0.00508
Neutron Generation Time, ℓ^* (μsec)		
HZP-BOL, ARO	27.2	18.1
HZP-EOL, ARO	31.8	19.7
Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{N*}$		
HZP-BOL, ARO, No Xe		1.64
HFP-BOL, ARO, No Xe		1.46
HFP-BOL, ARO, Eq. Xe		1.41
HFP-BOL, Bank 7 In, Eq. Xe		1.59
HFP-EOL, ARO		1.35
HFP-EOL, Bank 7 In		1.53

* Best estimate, no uncertainties or bias

TABLE 2

Total Planar Radial Peaking Factors
(Including Bias and Uncertainty)

Values of F_r	Cycle 3[1]	Cycle 4
Unrodded Region	1.60	1.59
Bank 7 Inserted	1.81	1.74
Values of F_{xy}		
Unrodded Region	1.58	1.60
Bank 7 Inserted	1.82	1.74

REFERENCES

1. Letter, Counsil to Reid, Millstone Nuclear Power Station, Unit No. 2, Proposed License Amendment, Power Uprating, Docket No. 50336, February 12, 1979.

2. Letter, Counsil to Grier, Millstone Nuclear Power Station Unit No. 2, Startup Testing Report, Docket No. 50-336, September 7, 1979.

4. Discuss the effects of using a different DNBR correlation for Cycle 4 transient analysis than was used for Cycle 3.

Response:

For the Cycle 3 analysis which uses the CE-1 correlation, DNB is not predicted to occur if a DNBR of 1.19 is met. For Cycle 4 analysis which uses the W-3 correlation, DNB is not predicted to occur if a DNBR of 1.30 is met. Since two different DNB correlations (both approved by the NRC) have been used, a direct comparison of the absolute DNBR values is not valid. The Cycle 4 analyses has shown that the effects (if any) of using a different DNB correlation for Cycle 4 than was used in Cycle 3 are negligible. That is, the conclusions drawn for the Cycle 4 analyses (e.g. DNB will not occur for Condition II transients) are the same as that determined for Cycle 3.

5. For the CEA ejection accident at both HFP and HZP, how many fuel rods go into DNB and what is the maximum RCS pressure attained?

RESPONSE

Since the CEA ejection transient is a very short power spike event, the fuel limits are best defined in terms of peak fuel enthalpy, rather than DNB ratio. This is consistent with the criterion set forth in Regulatory Guide 1.77. The CEA ejection analysis results presented in the Basic Safety Report and in the subsequent Reload Safety Evaluation Report for Cycle 4 indicate that the fuel limit for the transient is not exceeded. In fact, these results are less limiting than the results reported for Cycles 2 and 3. Therefore, the number of rods in DNB would be expected to be less in Cycle 4 than in previous cycles.

The RCS pressure spike resulting from the rod ejection is of no concern unless hot spot energy depositions in excess of 400 cal/gm are calculated, above which a pressure pulse could be postulated. These conclusions are the results of extensive TREAT and SPERT experiments. Therefore, the RCS pressure was not determined since the maximum hot spot heat deposition for this event was calculated to be less than or equal to 172 cal/gm as reported in Reference (1).

6. Previous cycle (Cycle 3) parameters assumed in the CEA drop analysis are identical to those assumed for Cycle 4 except for the more negative moderator temperature coefficient in Cycle 4. The minimum DNBR attained in the previous cycle analysis using the CE-1 correlation was 1.21. Since the maximum negative moderator temperature coefficient results in the minimum transient DNBR, why is the minimum DNBR obtained in the Cycle 4 analysis higher than that obtained in the Cycle 3 analysis? Also, since the EOC moderator temperature coefficient is much more negative than the BOC coefficient, why is it not used in the CEA drop analysis?

Response:

All DNB ratios reported by Westinghouse are based upon the W-3 correlation, and are not directly comparable to any Cycle 3 DNB ratios, which are based upon the CE-1 correlation. Further discussion on this is given in the response to question 4.

The minimum DNB ratio attained during the CEA drop accident is not very sensitive to the moderator temperature coefficient. A more negative moderator temperature coefficient would tend to return the core to full power with a smaller reduction in core inlet temperature. Since only manual rod control is available at Millstone, there would be no power overshoot due to automatic rod motion in response to the dropping of a CEA. The DNB ratio would not fall below its value at initial operating conditions with the dropped rod.

Since the EOC moderator temperature coefficient is more negative than the BOC value, the EOC moderator temperature coefficient was assumed for the CEA drop analysis in the BSR.

7. The PALADON computer code has not been approved by the staff for three-dimensional calculations. Provide a description of the types of calculations performed by PALADON for the Cycle 4 analysis.

RESPONSE

PALADON two-dimensional calculations were used for the following Cycle 4 analyses:

(a) Cold shutdown and refueling boron concentrations.

(b) Dropped rod power distribution.

These types of applications of PALADON have been approved by the staff per the "Safety Evaluation of WCAP-9485", J. F. Stolz to T. M. Anderson, dated September 12, 1979.

8. Please submit values for the following variables that were not provided in the Millstone 2 small-break LOCA ECCS performance results.

   a. Hot rod

      (1) differential pressure at time of rupture

      (2) temperature at time of rupture

      (3) axial distribution of circumferential strain

   b. Hot assembly

      (1) time of blockage

      (2) differential pressure at time of blockage

      (3) temperature at time of blockage

      (4) axial distribution of reduction-in-flow area

<u>RESPONSE</u>

In discussion with the NRC Staff at a meeting at your offices in Bethesda on March 18, 1980 and as documented in Reference (7), it was NNECO's understanding that a detailed review of the Westinghouse small break LOCA model for Millstone Unit No. 2 would not be made prior to model changes required by the Staff as a result of the TMI-2 accident. If the above understanding remains correct, the relevance of this question to the Cycle 4 reload is unclear. The Cycle 3 small break LOCA results are expected to serve as the basis for the Staff's evaluation. If this understanding is incorrect, NNECO respectfully requests clarification in this regard.

Nonetheless, responses are provided as follows:

(a) 1. Differential pressure at time of rupture: 457 psi

    2. Temperature at time of rupture: 1633^{O}F

    3. Axial distribution of circumferential strain:

Location*	Strain at Burst, $(\frac{\Delta D}{D})$
6.5 and below	$<10^{-6}$
7.63	2×10^{-4}
8.544	0.021
8.886	0.055
9.4 and above	0.1

(b) 1. Time of rupture: 1035 seconds

2. Differential pressure at time of rupture: 501 psi

3. Temperature at time of rupture: 1614^{O}F

4. Axial distribution of reduction - in-flow-area: As described in section 2.0 of Reference 1, the small break analysis is performed with a model which conservatively addressed flow in hot rod heat-up calculations by using the steam flow rate associated with an unblocked average rod. If a consideration of blockage effects were combined with use of the steam flows that encompass the hot rod, the increase in steam flow rate would result in a PCT reduction from the Millstone 2 Cycle 4 related small break ECCS analyses.

*Distance (feet) above bottom core

Ref (1): Addendum to WCAP-9528, Oct. 1979

9. Please submit values for the following variables that were not provided in the MP2 large break LOCA ECCS performance results.

   a. Hot rod
      (1) differential pressure at time of rupture
      (2) temperature at time of rupture
      (3) axial distribution of circumferential strain
      (4) time of peak cladding temperature

   b. Hot assembly
      (1) time of blockage
      (2) differential pressure at time of blockage
      (3) temperature at time of blockage
      (4) axial distribution of reduction-in-flow area

Response:

Information below is provided for the limiting break ECCS analysis submitted in support of the cycle 4 reload for Millstone 2 (i.e. C_D = 0.6 DECLG break);

(a) 1. Differential Pressure at time of rupture: 731 psi
    2. Temperature at time of rupture: 1648^{o}F
    3. Axial distribution of circumferential strain:

Location*	Strain at Burst, $(\frac{\Delta D}{D})$
2.848 and below	$\leq 3 \times 10^{-3}$
4.0	0.0473
4.5	0.1
Between 4.5 and 7.0	0.1
7.5	0.0483
8.0	0.0195
8.544 and above	$\leq 5 \times 10^{-3}$

    4. Time of peak cladding temperature: 162.6 seconds

(b) Burst is not predicted for the hot assembly rod in the Cycle 4 reload large break ECCS analysis for Millstone Unit 2.

*Distance (feet) above bottom of core

10. The NRC staff has been generically evaluating three materials models that are used in ECCS evaluation models. Those models are cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. Subsequent to Westinghouse submittals and your application of WCAP-9528, "ECCS Evaluation Model for Westinghouse Fuel Reloads of Combustion Engineering NSSS," and its addendum, we have (a) met and discussed our review with Westinghouse and other industry representatives (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis, and (c) required fuel vendors and licensees to confirm that the plants would continue to be in conformance with the ECCS criteria of 10 CFR 50.46 if the materials models of NUREG-0630 were substituted for those models of their ECCS evaluation models and certain other compensatory model changes were allowed.

The Westinghouse materials that are described in WCAP-9528 are virtually the same as those used in prior Westinghouse ECCS evaluation models, and they were evaluated in NUREG-0630. Small differences are attributable to modifications that were made to reflect the geometrical differences in fuel designs for the Millstone 2 plant. Therefore, until we have completed our materials model review, we will require plant analyses performed with the ECCS evaluation model as described in WCAP-9528 to be accompanied by supplemental analyses to be performed with the materials models of NUREG-0630. Therefore we request that NNECO submit a sample calculation as described above.

RESPONSE

The possible penalties for fuel rod models proposed by the NRC Staff in NUREG-0630 has been considered and the following information is provided.

A. Evaluation of the potential impact of using fuel rod models presented in draft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for Millstone 2 Reload, Cycle 4.

This evaluation is based on the limiting break LOCA analysis identified as follows:

BREAK TYPE - DOUBLE ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT C_D=0.6, pump suction

WESTINGHOUSE ECCS EVALUATION MODEL VERSION CE NSSS

CORE PEAKING FACTOR *2.464

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 1748 °F = PCT_B

ELEVATION - 5.70 Feet.

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 2111 °F = PCT_N

ELEVATION - 7.5 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION 0.13 Percent
MAXIMUM CLAD STRAIN AT THIS ELEVATION - 4.83 Percent

Maximum temperature for this non-burst node occurs when the core reflood rate is (GREATER) than 1.0 inch per second and reflood heat transfer is based on the (FLECHT) correlation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - NA Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - NA Percent

1. BURST NODE

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in terms of the change in the peaking factor limit (FQ) required to maintain a peak clad temperature (PCT) of 2200°F and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above 2200.°F, individual effects (such as ΔPCT due to changes in several fuel rod models) indicated here may not accurately apply over large ranges,

*(Peak kw/ft ÷ Average kw/ft)

but a simultaneous change in FQ which causes in the neighborhood of 2200.°F justifies use of this evaluation procedure.

From NS-TMA-2174:

For the Burst Node of the clad:

- $\overline{0.01\ \Delta FQ} \to \sim 150^{\circ}F$ BURST NODE ΔPCT
- Use of the NRC burst model and the revised Westinghouse burst model could require an FQ reduction of 0.027
- The maximum estimated impact of using the NRC strain model is a required FQ reduction of 0.03.

Therefore, the maximum penalty for the Hot Rod burst node is:

$$\Delta PCT_1 = (0.027 + .03)\ (150^{\circ}F/.01) = 855^{\circ}F$$

Margin to the 2200°F limit is:

$$\Delta PCT_2 = 2200.^{\circ}F - PCT_B = \underline{452}\ ^{\circ}F$$

The FQ reduction required to maintain the 2200°F clad temperature limit is:

$$\Delta FQ_B = (\Delta PCT_1 - \Delta PCT_2)\ \left(\frac{.01\ \Delta FQ}{150^{\circ}F}\right)$$

$$= (\underline{855} - \underline{452})\ \left(\frac{.01}{150}\right)$$

$$= \underline{.0269}$$ (but not less than zero).

2. NON-BURST NODE

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspects of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which burst is calculated. Three sets of LOCA analysis results were studied to establish an acceptable sensitivity to apply generically in this evaluation. The possible PCT increase resulting from a change in strain (in the Hot Rod) is +20.°F per percent decrease in strain at the maximum clad temperature locations. Since the reactor coolant system blowdown phase of the accident is not changed by the use of NRC fuel rod models, the maximum decrease in clad strain that must be considered here is the difference between the "maximum clad strain" and the "clad strain at the end of RCS blowdown" indicated above.

Therefore:

$$\Delta PCT_3 = \left(\frac{20^{\circ}F}{.01\ \text{strain}}\right)\ (\text{MAX STRAIN - BLOWDOWN STRAIN})$$

$$= \left(\frac{20}{.01}\right)\ (4.83 - .13) \times 10^{-2}$$

$$= 94^{\circ}F$$

The second aspect of the analysis that can increase PCT is the flow blockage calculated. Since the greatest value of blockage indicated by the NRC blockage model is 75 percent, the maximum PCT increase can be estimated by assuming that the current level of blockage in the analysis (indicated above) is raised to 75 percent and then applying an appropriate sensitivity formula shown in NS-TMA-2174.

Therefore,

$$\Delta PCT_4 = 1.25^{\circ}F\ (50 - \text{PERCENT CURRENT BLOCKAGE}) + 2.36^{\circ}F\ (75-50)$$

$$= 1.25\ (50 - \_\_\_\_)\ + 2.36\ (75-50)$$

$$= \_\_\_\_\ ^{\circ}F$$

If PCT_N occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 94^{\circ}$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200^{\circ}F - PCT_N = 89^{\circ}$$

The FQ reduction required to maintain this 2200°F clad temperature limit is (from NS-TMA-2174)

$$\Delta FQ_N = \underbrace{(\Delta PCT_5 - \Delta PCT_6)}^{94 - 89}\ \left(\frac{.01 \Delta FQ}{10^{\circ}F\ \Delta PCT}\right)$$

ΔFQ_N = -.005 but not less than zero.

The peaking factor reduction required to maintain the 2200°F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_N,

or; $\Delta FQ_{PENALTY} = 0.0269$

B. The effect on ECCS analysis results of using improved, more representative data has been assessed in relation to the ECCS analysis performed and submitted for the cycle 4 reload of the Millstone 2 plant. It has been determined that the margin involved in the conservatism of input parameters is more than adequate to offset potential burst-blockage model impacts. Specifically, design value fuel pellet temperatures were assumed for the Millstone 2 ECCS analysis involving Westinghouse fuel. Fuel parameters specific for cycle 4 confirm the existence of additional margin (33°F) compared to the values utilized in the analysis.

Previous licensing credits applied to the W evaluation model analysis have resulted in a minimum F_Q increment of 0.07 for each 85°F reduction in pellet temperature. Therefore, incorporating the cycle-4 specific fuel information would result in a cycle 4 margin of 0.0271 in F_Q for the 33°F margin in the pellet temperature parameter for the cycle 4 Millstone 2 fuel. Hence, consideration of pellet temperature-related input confirms that adequate margin exists in the ECCS analysis submittal to preclude any F_Q or peak kw/ft adjustments associated with burst-blockage considerations.

C. The peaking factor limit adjustment required to justify plant operation for this burst-blockage issue is determined as the appropriate ΔF_Q credit identified in section (B) above, minus the $\Delta FQ_{PENALTY}$ calculated in section (A) above (but not greater than zero):

$$F_Q \text{ ADJUSTMENT} = 0.0271 - 0.0269 \sim 0$$

This evaluation demonstrates that a conservative assessment of those penalties is compensated for by available improvements in the ECCS analysis already provided to the Staff. The procedure utilized to perform the analysis is deemed appropriate and suitably conservative and provided adequate supplementary material until final resolution of the overall fuel rod model concern is achieved.

The format is similar to evaluations already provided to the NRC to support licensing of Westinghouse-NSSS operating plants. Credits specific to the Cycle 4 Millstone 2 reload have been developed.