NORTHEAST UTILITIES

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August 14, 1980

Docket No. 50-336
A01085

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

References: (1) R. A. Clark letter to W. G. Counsil dated June 20, 1980.
(2) W. G. Counsil letter to R. A. Clark dated July 7, 1980.
(3) W. G. Counsil letter to R. A. Clark dated July 22, 1980.
(4) R. A. Wiesemann letter to H. R. Denton dated February 29, 1980.
(5) R. A. Wiesemann letter to D. G. Eisenhut dated July 27, 1976.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Response to Questions on Cycle 4 Basic Safety Report

In Reference (1), the NRC Staff requested Northeast Nuclear Energy Company (NNECO) to provide additional information regarding fuel design and physics calculations to continue review of the Basic Safety Report (BSR). In References (2) and (3), NNECO responded to questions identified in Reference (1), and indicated that this information was non-proprietary.

It has subsequently been determined that the information docketed in References (2) and (3) is proprietary to Westinghouse Electric Corporation and is respectfully requested to be withdrawan from the Public Document Room. The non-proprietary version of the responses is provided as Attachment 1.

Due to the proprietary nature of portions of the material contained in References (2) and (3), NNECO requests that it be withheld from public disclosure in accordance with the provisions of 10CFR2.790 and that this material be safeguarded. The reasons for the classification of this

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material as proprietary are delineated in the affidavits previously docketed by References (4) and (5).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

Attachment

W. F. Fee

By: W. F. Fee
Executive Vice President

DOCKET NO. 50-336

ATTACHMENT 1

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

RESPONSES TO NRC QUESTIONS ON BASIC SAFETY REPORT

NON-PROPRIETARY

AUGUST, 1980

Responses to NRC Questions on Westinghouse
Basic Safety Report for Millstone Unit 2

1. Please submit page 5-14 which is missing from the BSR.

   RESPONSE: Please find enclosed with this submittal copies of the missing page.

2. Coolant pressure drop calculations for the Westinghouse fuel assembly design indicates a matching of the overall pressure drop with that for the original Combustion Engineering Millstone 2 fuel assembly design. However, at each axial elevation the pressure drops do not match up between the Westinghouse and/Combustion Engineering fuel designs. The largest variation in pressure drops for the two designs occurs at the lower nozzle where the Westinghouse design has a higher pressure-loss coefficient. This variation in pressure drop will result in an inlet flow maldistribution with less direct flow through the Westinghouse bottom nozzle. The BSR should provide justification as to why the resulting cross flow downstream of the bottom nozzle will not produce an unacceptable degree of fretting wear at sites where spacer grid springs and dimples contact fuel rods.

   RESPONSE: As stated in Section 3.3 and shown in Table 3.1 of the BSR, the Westinghouse fuel assembly design is hydraulically compatible with the Combustion Engineering Millstone 2 fuel assembly design. The geometric similarity of all the assembly components and subsequent testing have assured that the overall assembly pressure drops as well as the pressure drops at each axial elevation are matched for the two designs.

   The lower nozzle loss coefficient mismatch which was a concern at the Feb. 14, 1977 meeting (Reference 1) was resolved by a re-design of the lower nozzle and subsequent testing to determine the effects of rods lifted and not lifted. This nozzle re-design and test results were presented at the January 26, 1979 meeting (Reference 2) and showed that the W lower nozzle was hydraulically compatible with the CE lower nozzle. Thus there will be no inlet flow maldistribution between the two designs.

References

(1) M. M. Mendonca, NRC, Memorandum to R. L. Baer, NRC, Subject: "Westinghouse Design and Testing Program for Millstone 2 Reload Number 1 Meeting - February 14, 1977", April 19, 1977.

(2) W. G. Counsil, Northeast Nuclear Energy Company, letter to R. Reid, NRC, "Millstone Nuclear Power Station, Unit No. 2 Cycle 4 Reload Regarding January 26, 1979 Meeting", March 21, 1979.

3. The Westinghouse fuel assembly design has 4 holddown springs, while the original Combustion Engineering design has 5 springs. Discuss the differences in the static and dynamic response of each fuel assembly design. The raised pad on the center of the top nozzle orifice plate prevents the Westinghouse holddown springs from being compressed solid. Does this pad limit the axial distance that the Westinghouse fuel assemblies can grow relative to that of the original Combustion Engineering fuel assemblies? Will the spacer grids of the two fuel assembly designs always line up? What is the safety significance if grid-to-grid alignment cannot be assured (i.e., will there be neutronic anomalies, will assembly peripheral fuel rods be punctured)?

RESPONSE: The W fuel assembly holddown springs are designed to prevent lift off of the fuel assembly from the bottom core plate during normal operation. This design requirement is consistent with the CE fuel assembly design of the holddown spring as discussed in section 3.3.1.4 of the Millstone Unit 2 FSAR. Since both the CE and W fuel assembly holddown loads are similar, the fuel assembly spring rates are judged to be similar, thus the static and dynamic response of each fuel assembly should not differ significantly.

The raised pad at the center of the W top nozzle orifice plate will not interfere with the free growth of the fuel assembly. It is located such that it contacts the holddown flower prior to the springs being compressed solid and prevents the nozzle extensions from topping out in the blind holes in the upper core plate. This promotes uniform loading of the guide tubes during any hypothetical accident which could cause the fuel assembly to lift.

Since the raised pad does not prevent fuel assembly growth, there will always be grid to grid overlap between W and CE fuel assemblies even when considering irradiation growth.

4. The BSR states that cladding flattening is precluded during the projected exposure of the fuel. Provide the minimum time to collapse as calculated with the COLLAP code. What is the design maximum value of fuel assembly burnup?

   RESPONSE: The clad flattening time is predicted to be $\geq$ 50000 EFPH for region 6 fuel using the current Westinghouse evaluation model.[1] Region 6 fuel, comprised of 6-1 and 6-2, has a projected residence time through 3 cycles of ~27000 EFPH.

5. What is the calculated minimum shoulder gap which allows for differential growth between fuel rods and the fuel assembly? Provide the two Zircaloy growth correlations used in this calculation and describe or provide the data base from which these correlations were determined. How were the growth correlations combined with (a) fabrication tolerances, (b) differential thermal strains of the fuel assembly and reactor internals, and (c) elastic compression and creep of the guide thimble tubes? For steady-state operation, at what axially-averaged assembly burnup will interference result in rod bow?

   RESPONSE: To insure that no axial interference between the fuel rod and the assembly nozzles can occur, the W Millstone Unit 2 fuel rod was designed so that the minimum room temperature clearance value would be equal to or greater than [ ]^{a,c} percent times the fuel rod length [ ]^{a,c}. Extensive operating experience with other fuel assemblies using the same materials and having equivalent fuel duty has shown that this design value for clearance is conservative with respect to temperature and irradiation induced length changes of the fuel rod and fuel assembly. Therefore, under normal steady-state operation, there will be no interference of fuel rods since the growth allowance will preclude rod bow.

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(1) [illegible]

6. The NRC staff has not commenced the review of the Westinghouse generic topical report WCAP-8691, Revision 1, "Fuel Rod Bowing Evaluation", which is referenced in the BSR. Specifically, the BSR uses a formula from WCAP-8691 that projects anticipated rod bow magnitudes due solely to geometrical changes in the fuel rod thickness and diameter and spacer grid span length. This formula has been somewhat controversial and has not been accepted by the staff. Therefore, we will require that the degree of rod bowing in the Westinghouse reload fuel be calculated with the existing approved method, which is relatively more conservative. In spite of this additional conservatism, however, we do not calculate a need for a DNBR penalty until an assembly burnup of 36,300 MWD/MTU is attained at which exposure the 50% gap closure value is reached. We require that Westinghouse confirm our calculations and verify that no other changes in fuel design variables (i.e., grid spring preload, degree of cladding cold work, etc.) are significant to the rod bowing extrapolation for the Millstone, Unit 2 reload fuel.

RESPONSE: The degree of fuel rod bowing in the Westinghouse reload fuel has been recalculated with the existing, more conservative approved method:

$$\frac{L_W\,(M.S.II)/I_W\,(M.S.II)}{L_{15x15}\,/\,I_{15x15}} = 0.59$$

where L = span length
I = cross-sectional moment of inertia

The average burnup at which a gap closure of 50% is attained is 32,000 MWD/MTU. By the time the fuel attains a burnup of 32,000 MWD/MTU, it is not capable of achieving limiting peaking factors due to the decrease in fissionable isotopes and the buildup of fission product inventory. This physical burndown effect is greater than the rod bowing effects which would be calculated at those burnups. Therefore the effect of rod bow need not be considered in the analysis of the Millstone II core.

The fuel design variables were selected according to standard Westinghouse design practice. Therefore, the rod bowing extrapolation method is valid.

7. The Combustion Engineering supplied fuel for Millstone, Unit 2 was designed according to a specific set of Specified Acceptable Fuel Design Limits (SAFDLs). Please list all of the Westinghouse SAFDLs for the Millstone reload fuel and provide the basis for omissions or additions to the original Combustion Engineering set of SAFDLs.

RESPONSE: There are only two Specified Acceptable Fuel Design Limits (SAFDL's) covered by the Reactor Protection System (RPS), and these are discussed in Chapter 6 of the Basic Safety Report (BSR). These SAFDL's are:

1) The peak linear heat rate must be below that which would cause incipient fuel centerline melting. The melting point limit is conservatively taken as 4700°F to bound the effects of fuel burnup and uncertainties in the melting point.

2) The DNB thermal limits must not be exceeded.

The SAFDL's as defined by Westinghouse are equivalent, or more conservative, when compared to the CE SAFDL's as given in Chapter 1 of CENPD-199:

a) The reactor fuel shall not experience centerline melt.

b) The departure from nucleate boiling ratio (W-3 DNBR) shall have a minimum allowable limit of 1.3.

Further discussion of fuel design criteria may be found in Chapter 2 of the BSR and Chapter 3 of the FSAR.

8. Some of the accident anlayses described in the BSR were performed with the computer codes FACTRAN (WCAP-7908, "FACTRAN, A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod") and LOFTRAN (WCAP-7907, "LOFTRAN Code Description"). Our review of these topicals has progressed to the point that there is reasonable assurance that the conclusions based on these analyses will not be appreciably altered by completion of the analytical review, and therefore that there will be no effect on the decision to issue a license amendment. If the final approval of these topical reports indicates that any revisions to th [illegible] will be required to implement the results of such changes.

RESPONSE: We do not expect that any revised analyses will be required as a result of your review of FACTRAN or LOFTRAN.

9. Please either reference or provide a thorough description of the Westinghouse Computer Analyses Code (WECAN), which was used to perform the stress analyses of fuel assembly components.

RESPONSE: A description and benchmark problem solutions have been submitted to the NRC via Reference 1 below.

Ref. 1 - WCAP-8929, "Benchmark Problem Solutions Employed for Verification of the WECAN Computer Program", April 1977.

10. Comparisons of power peaking in fuel pins adjacent to CEA water holes using TURTLE (diffusion theory) and KENO (Monte Carlo) have shown an underprediction by diffusion theory, as expected. Please provide additional information, such as comparisons between KENO calculations and experimental measurements of water hole power peaking, to justify the KENO calculational uncertainty used.

RESPONSE: The total water hole peaking factor bias to be used in INCA can be calculated from (TURTLE-KENO) plus (KENO-experiment) differences which is equivalent to the bias (TURTLE-experiment). By inference, the difference between TURTLE and water hole experiments was calculated using INCA results. For purposes of licensing TURTLE for use with large water hole lattices, the INCA comparisons described below justify a water hole peaking factor bias of 2.8% to be used in INCA for measured peaking factors.

The total water hole peaking factor bias (TURTLE-experiment) was calculated from a comparison of INCA and TURTLE values of the ratio, hot rod to assembly average power, in cycles 1, 2 and 3. This comparison provides the water hole bias because:

a) The ratio, hot rod to assembly average power, from INCA is the same as the hot pin relative power predicted by the design code (PDQ) used for INCA input in Cycles 1, 2 and 3.

b) The hot pin power always occurs next to a water hole.

c) The water hole peaking factor bias used in INCA for Cycle 3 is 4.8%[1] which results from extensive comparisons between PDQ and water hole experiments.

d) The total water hole peaking factor bias to be used in INCA with TURTLE input is bounded by correcting the Cycle 3 bias of 4.8% by the bias between TURTLE and PDQ hot pin powers.

Comparisons between TURTLE and INCA peak to average rod power for six INCA maps in cycles 1 and 2 are given in the BSR and results for two Cycle 3 maps are shown in Figures 2 - 5. The results for all 3 cycles are given in Figure 1 and indicate a bias of 2% between TURTLE and PDQ.

The total water hole peaking factor bias to be used in INCA with TURTLE is thus 4.8 - 2.0 = 2.8%.

11. The fuel rod support grid for Cycle 4 will be Inconel-718 rather than Zircaloy-4 as used in Cycle 3. What are the effects of this material charge on power distributions and other physics parameters?

RESPONSE: The effects of the nuclear and thermal expansion properties of Inconel-718 grids in Westinghouse - supplied fuel assemblies and Zircaloy-4 grids in CE supplied fuel assemblies were considered in the evaluation of physics parameters (e.g., reactivity coefficients) for the Millstone 2, Cycle 4 core.

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(1) CEN-88(N)-NP, Increased Water Hole Peaking in Operating Reactors (Millstone 2), March 30, 1978.

Figure 1. Comparison of PDQ and TURTLE Assembly Peak to Average Ratios

a, c

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SOURCES OF DATA

1) Cycles 1 and 2 - WCAP-9660 Figures 4-18 through 4-23

2) Cycle 3 - Figures 2-5 (attached)

Figure 2. CE INCA

Cycle 3 3250 MWD/MTU

KEY: ASSY ID OCTANT POS
FQ
FXY
FR
FZ
RPD BASED ONLY ON N. OF RODS

	V04 33	W04 34
FQ	1.117	1.314
FXY	1.016	1.193
FR	1.001	1.161
FZ	1.116	1.135
RPD	0.849	0.714

	T05 29	V05 30	W05 31	X05 32
FQ	1.183	1.232	1.399	1.447
FXY	1.102	1.129	1.272	1.314
FR	1.112	1.135	1.259	1.274
FZ	1.086	1.105	1.117	1.139
RPD	0.937	0.931	1.048	0.732

	S06 24	T06 25	V06 26	W06 27	X06 28
FQ	0.944	1.430	1.610	1.427	1.626
FXY	0.871	1.314	1.474	1.295	1.464
FR	0.874	1.310	1.460	1.281	1.414
FZ	1.100	1.097	1.105	1.118	1.152
RPD	0.773	1.145	1.281	1.132	1.000

	R07 18	S07 19	T07 20	V07 21	W07 22	X07 23	Y08 17
FQ	1.101	1.254	1.376	1.124	1.168	1.613	1.491
FXY	1.029	1.166	1.280	1.058	1.079	1.465	1.353
FR	1.020	1.148	1.266	1.035	1.096	1.417	1.304
FZ	1.089	1.109	1.096	1.083	1.100	1.136	1.147
RPD	0.901	0.981	1.090	0.901	0.879	1.146	0.769

	N09 10	R09 11	S09 12	T09 13	V09 14	W09 15	X09 16	Y10 09
FQ	1.605	1.486	1.076	1.605	1.476	1.241	1.422	1.554
FXY	1.487	1.383	1.025	1.472	1.362	1.141	1.294	1.412
FR	1.464	1.362	1.010	1.462	1.346	1.133	1.277	1.357
FZ	1.098	1.095	1.087	1.100	1.098	1.104	1.120	1.146
RPD	1.252	1.168	0.888	1.266	1.170	0,925	1.113	0.459

	L11 1	N11 2	R11 3	S11 4	T11 5	V11 6	W11 7	X11 8
FQ	0.864	1.092	1.134	1.199	1.233	1.126	1.561	1.182
FXY	0.796	1.015	1.063	1.140	1.169	1.075	1.423	1.075
FR	0.796	1.009	1.058	1.111	1.174	1.084	1.414	1.057
FZ	1.098	1.090	1.083	1.083	1.076	1.033	1.107	1.128
RPD	0.722	0.888	0.899	0.991	0.936	0.914	1.241	0.942

Fig. 3 - CE INCA CYCLE 3
5850 MWD/MTU

KEY: ASSY ID OCTANT POS
FQ
FXY
FR
FZ
RPD

$\frac{F_R}{RPD}$ = PEAK TO AVG.

NOTE: F_Q, F_{xy} and F_R include a 4.8% bias factor

	V04 33	W04 34
FQ	1.111	1.298
FXY	1.029	1.197
FR	1.015	1.169
FZ	1.096	1.113
RPD	0.862	0.726

	T05 29	V05 30	W05 31	X05 32
FQ	1.169	1.202	1.353	1.401
FXY	1.095	1.120	1.249	1.289
FR	1.107	1.126	1.242	1.260
FZ	1.074	1.088	1.096	1.116
RPD	0.944	0.937	1.043	0.733

	S06 24	T06 25	V06 26	W06 27	X06 28
FQ	0.935	1.389	1.562	1.375	1.551
FXY	0.890	1.294	1.449	1.264	1.437
FR	0.889	1.297	1.437	1.258	1.383
FZ	1.069	1.077	1.089	1.096	1.131
RPD	0.792	1.143	1.271	1.117	0.984

	R07 18	S07 19	T07 20	V07 21	W07 22	X07 23	Y08 17
FQ	1.091	1.223	1.334	1.113	1.134	1.548	1.428
FXY	1.030	1.155	1.256	1.062	1.061	1.423	1.312
FR	1.030	1.142	1.250	1.041	1.081	1.388	1.276
FZ	1.068	1.078	1.075	1.067	1.082	1.115	1.123
RPD	0.916	0.994	1.009	0.905	0.881	1.123	0.761

	N09 10	R09 11	S09 12	T09 13	V09 14	W09 15	X09 16	Y10 09
FQ	1.572	1.458	1.072	1.566	1.436	1.703	1.361	[illegible]
FXY	1.470	1.364	1.025	1.451	1.346	1.131	1.254	[illegible]
FR	1.463	1.359	1.019	1.438	1.333	1.128	1.246	[illegible]
FZ	1.075	1.077	1.068	1.090	1.079	1.077	[illegible]	1.123
RPD	1.261	1.175	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.945

	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	W11 7	X11 8
FQ	0.878	1.083	1.132	1.190	1.208	1.101	1.513	1.141
FXY	0.830	1.070	1.067	1.140	1.146	1.060	1.398	1.052
FR	0.826	1.073	1.071	1.120	1.156	1.070	1.392	1.041
FZ	1.064	1.065	1.068	1.066	1.071	1.067	1.099	1.105
RPD	0.751	0.906	0.917	1.007	0.938	0.918	1.232	0.934

Fig. 4 — AVERAGE ASSEMBLY POWER, PEAK ASSEMBLY POWER

a, c

Fig.5- AVERAGE ASSEMBLY POWER, PEAK ASSEMBLY POWER

a,c

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Calculations of $F_Q(Z)$ include a multiplicative factor, applied to the axial peaking factors, to account for axial inhomogeneities introduced by assembly grids. The inclusion of the grid multiplicative factor bounds the inhomogeneities due to either Zircaloy or Inconel grids.

12. Power distributions calculated by TURTLE appear to underpredict the power in the peripheral assemblies while overpredicting the power in the center assemblies. In view of the large errors in predicting CEA worth near the core periphery due to these power distribution inaccuracies, justify the use of TURTLE without some type of baffle correction scheme.

<u>RESPONSE</u>: As noted in the BSR, CEA-3 control bank worth on the periphery of the core was underpredicted relative to measurement in Cycles 1, 2 and 3. This results from a slight underprediction of the power in the peripheral assemblies.

The BOL, HZP control worths were calculated again with a baffle correction of [ ]^{a,c} and are shown in Table 1 along with the original results. As seen, the CEA-3 agreement is improved by the baffle correction. The total control worth, 2-7, remains virtually unchanged as expected. The baffle correction will be used for Cycle 4 design.

Table 1

Effect of Baffle Correction on Cycle 3 CEA-3 Worth

		% Difference*	
CEA Worth, %$\Delta\rho$	Measured	No Corr.	Baffle Corr.
7	0.64	0.31	-1.56
6	0.25	-16.80	-12.00
5	0.17	-24.70	-13.53
4	0.88	- 8.64	-10.80
3	0.67	7.61	3.23
2	1.15	0.70	-0.35
Center (7-1)	0.03	-27.90	-21.67
Sequential Worth, 2-7	3.76	- 3.21	-3.72

*(Measured-predicted)/measured

13. The CEA drop analysis was performed without automatic rod control (CEA motion inhibit) or turbine load reduction. Is this the operational plan for Cycle 4.

RESPONSE: CEA motion inhibit and turbine load reduction tend to mitigate the consequences of a CEA drop. Therefore, no credit was taken for these functions in the CEA drop analysis. Turbine load reduction will not be operational; but the CEA motion inhibits will be available during Cycle 4.

14. The parameters used in the analysis of the CEA ejection accident appear to be representative of Westinghouse cores and differ from the previous Millstone, Unit 2 fuel supplier in several areas such as ejected rod worths, ejection time, delayed neutron fraction, feedback reactivity weighting, and power peaking (F_q). Please provide a comparison between the Cycle 3 and Cycle 4 values of these rod ejection initial assumptions and discuss the reasons for and effects of any differences.

RESPONSE: Comparison of CEA ejection accident parameters are given in Table 1 for Cycles 3 and 4.

In Cycle 3, the ejected rod worth is larger than the Cycle 4 value at HFP; at HZP, the values are the same. The difference at HFP is probably due to the assumption used for control bank insertion prior to ejection. As explained in Ref. 1, the method used by Westinghouse in Cycle 4 is to assume that the rod is ejected from control bank CEA-7 [ ].$^{(a,c)}$

The delayed neutron fraction used in Cycles 3 and 4 are the same.

The F_Q after ejection at HFP is slightly higher in Cycle 3 than in Cycle 4. Differences in radial and axial power distribution due to the burnup characteristics and location of the fuel in Cycle 4 account for the difference.

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1. WCAP-9272, March 1978, "Westinghouse Reload Safety Evaluation Methodology"

At HZP, the Cycle 4 F_Q after ejection is larger than the Cycle 3 value. Again, some of the difference can be accounted for by the change in fuel burnup and location between Cycles 3 and 4. Another important contribution is the axial shape assumed as a pre-condition. The Westinghouse methodology is to [

]^{a,c} which accounts for the large value of F_Q at HZP.

The CEA ejection time was assumed to be 0.1 seconds in Cycle 4. This value has no impact on the results compared to the ejection time of 0.05 seconds used in Cycle 3.

The feedback reactivity weighting used in Cycle 4 was applied only to the Doppler feedback, and conservatively accounts for the increased feedback due to the highly peaked power distribution following the CEA ejection. In addition, this weighting factor was applied to a conservative prediction of the zero to full power normal operation Doppler power defect of only 0.84% Δk. This methodology is described more fully in WCAP-7588 Rev. 1-A and in reference safety analysis reports. Although the Cycle 3 analysis employed a spatial Doppler feedback weighting factor, the value used was not reported for that cycle. However, values of 1.24 to 1.34 were reported for the full power cases and 1.94 to 2.52 for the zero power cases analyzed for Cycle 1 and reported in the Millstone 2 FSAR. These values are very close to the values used in the Cycle 4 analysis.

Table 1

Comparison of CEA Ejection Accident Parameters

Parameter	HFP Cycle 3(1)	HFP W Cycle 4(2)	HZP Cycle 3(1)	HZP W Cycle 4(3)
Ejected rod worth, Δk	0.29	0.17	0.65	0.65
Delayed neutron fraction	0.47	0.47	0.47	0.47
F_Q after ejection	5.83	5.70	14.5	18.8
Ejection time	.05	.1	.05	.1
Feedback reactivity weighting	(4)	1.30	(4)	2.50

1. Letter, Counsil to Reid, Millstone Unit No. 2 Power Uprating, Feb. 12, 1979
2. WCAP-9660, BSR, February, 1980
3. RSE, Millstone Unit 2, Cycle 4, May, 1980
4. Value not reported