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SUBJECT:
RESPONSE TO NRC LTR DTD 04/07/78... FORWARDING ANSWERS TO THE CYCLE 4 RELOAD QUESTIONS AND ADDL CONCERNS AS REQUESTED BY NRC.

PLANT NAME: THREE MILE ISLAND - UNIT 1

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1489 009

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METROPOLITAN EDISON COMPANY

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April 10, 1978
GQL 0658

Director of Nuclear Reactor Regulation
Attn: R. W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

Attached please find the answers to the Cycle 4 Reload questions and additional concerns identified in your letter of April 7, 1978.

Members of my staff remain available at your convenience to discuss any additional concerns that you may have in regard to this submittal.

Sincerely,

J. G. Herbein
Vice President-Generation

JGH:RJS:cjg

Attachment

A001
S
1/1

Question No. 1

Describe the changes to the CVCS necessary to use the feed-bleed mode of operation.

Response:

No changes to the TMI-1 Makeup and Purification System were required to support operation of the unit in the unrodded or feed-and-bleed mode beginning in Cycle 4. All B&W nuclear power plants are designed with the capability to conduct feed and bleed operations, independent of whether the core is operated in the rodded or unrodded mode. The design letdown flow-rate for all B&W 177 FA units is the same (140 gpm). In the rodded plants, feed and/or bleed operations are necessary to compensate the following reactivity changes:

- excess reactivity required for fuel burnup and fission product buildup over the fuel cycle (depletion effects).
- moderator temperature reactivity effects due to RCS coolant temperature changes at startup and shutdown.
- buildup of equilibrium xenon and samarium reactivity.
- boration to shutdown requirements specified by Technical Specificiations.
- deboration from shutdown or refueling concentration requirements during startup.

For operation in the unrodded mode, the required feed and bleed capabilities are the same as stated above with the addition of adjusting the RCS boron concentration to maintain the regulating control rods within specified maneuvering control bands during power level changes or load follow. Both boration and deboration are accomplished manually to keep the control rods in a predescribed operating band within the rod position limits of Technical Specifications. The maneuverability of the plant is then limited only by the ability of the waste processing system to handle the waste generated, as indicated in the attached Table 1.

TMI-1 has operated at end of cycles 1 and 3 in the "all rods out" or feed and bleed mode of operation.

1489 011

Response to Question No. 1

TABLE 1

INTERVAL BETWEEN PUSH-PULL LOAD CHANGES

Interval Days	Load Change Percentage	Last FPD Conc. (PPM)	Last Time In Life FPD	Last Time In Life %	Fluid to be Processed Gallons
1	70	900	30	10	14,500
2	70	450	186	60	29,000
3	70	320	220	71	42,500
4	70	250	244	79	58,000
5	70	230	250	81	72,500
1	50	510	168	54	14,500
2	50	265	239	77	29,000
3	50	185	264	85	42,500
4	50	150	279	90	58,000
			310	100	

1489 012

Question No. 2

The Tech. Spec. changes presented in the April 3, 1978 submittal appear to be based on cross-core shuffle of the fuel even though this refueling configuration is no longer being proposed. Describe in detail the effects of non-cross-core shuffle on the parameters contained in the January 9, 1978 submittal. Revise or verify all tables presented in the January submittal to reflect the effect of the additional cycle 3 burnup and the non-cross-core shuffle.

Response:

Met-Ed's submittal of April 3, 1978, addressed both the non-cross core shuffle scheme and the extended cycle 3. However, since the changes, due to the extended cycle 3, were more restrictive than the non-cross-core changes, the non-cross core changes were not indicated. For example, only a single setpoint required change as a result of the non-cross core shuffle scheme, i.e., the power imbalance negative limit at 102% power decreased from -30.80 (January 9, 1978 submittal) to -28.94 for operation from 0 to 125 ± 5 EFPD. The extended cycle 3 further decreased the power imbalance negative limit at 102% power to -23.40. Therefore, the change submitted April 3, 1978, was due to the extended cycle 3 and not to the non-cross core shuffle scheme. With respect to the Reload Report, which will be revised as indicated in the April submittal, the only changes resulting from the non-cross core shuffle are, 1) a revised Figure 3-1, Core Loading Diagram for TMI-1 Cycle 4 (submitted to NRC April 3, 1978); 2) a revised Figure 5-1 (attached); and 3) revised calculated nuclear peaks as follows:

	Radial-Local Peak	Margin to Ref. Design
Original Cycle 4 submittal (January 9, 1978)	1.637 (BOL) 1.421 (EOL)	8.2% (BOL) 20.3% (EOL)
Non-Cross Core Shuffle	1.596 (BOL) 1.407 (EOL)	10.5% (BOL) 21.1% (EOL)
Non-Cross Core Shuffle and extended Cycle 3 (287.1 EFPD) (To be included in revised Reload Report)	1.547 (BOL) 1.403 (EOL)	13.2% (BOL) 21.3% (EOL)

It should be noted that revised Figure 5-1, attached in response to this question, is superseded by revised Figure 5-1 attached in response to Question No. 5.

Figur -1. BOC (4 EFPD), Cycle 4 Two-Dimensional Relative Power Distribution — Full Power, Equilibrium Xenon, APSRs Inserted

	8	9	10	11	12	13	14	15
H	0.96	1.09	1.23	0.99	1.23	0.91	0.82	0.76
K		1.29	1.10	1.20	1.07	1.17	0.84	0.76
L			1.36		8 1.06	0.92	1.14	0.66
M				1.00	1.16	0.94	0.96	
N					1.02	1.11	0.68	
O						0.54		
P								
R								

X	Inserted Rod Group Number
X.XX	Relative Power Density

Question No. 3:

The beginning of cycle (BOC) boron concentration for cycle 4 reported in Table 1 of the April 3, 1978 submittal is less than that in the FSAR. Provide available operator response times for a boron dilution event occurring (1) during refueling, and (2) during startup, cold shutdown, hot standby, and power operation.

Response:

The BOC boron concentration for Cycle 4 is less than that given in the FSAR. This means the reactivity insertion rate due to a moderator dilution event at power is less for Cycle 4 than that given in the FSAR. Since the refueling boron concentration requirement will remain the same for Cycle 4, the FSAR analysis for the margin to critical for a shutdown condition remains valid.

The conditions of startup, cold shutdown and hot standby were not addressed in the FSAR and as such were not analyzed for Cycle 4 consistent with licensing by comparison to the FSAR.

Question No. 4

Tech. Spec. Change Request No. 75, dated March 1 , 1978, is for a change to allow a 4% uncertainty between the excore measured power and the power obtained by a plant heat balance. In view of the assumed 2% error in measured power required to be used in accident and transi t analyses, explain how the 4% uncertainty has been accounted for in the accident analyses and the protection system setpoints. If the additional uncertainty in power has not been accounted for in the accident analyses, provide new analyses, including ECCS, which properly reflect the additional 2% uncertainty.

Response:

The 2% heat balance error assumed in the Safety Analysis is retained in the setting of the Tech. Spec. setpoints. The 4% neutron power measurement error is also accounted for in the Safety Analysis. The following break down of assumed errors is presented to further clarify this issue:

112%	Safety Analysis Setpoint
- 2%	Heat Balance Error
- 4%	Neutron Power Measurement Error
-.5%	Bistable Setting Error
105.5%	Tech. Spec. Setpoint Value

Tech. Spec. Change Request No. 75 was submitted to account for the full 4% neutron power measurement error accounted for in the Safety Analysis. Of this 4% error, 2% is for steady-state measurement and 2% for transient effects. Part of Change Request 75 was a daily check of the power measurement, requiring a heat balance calibration whenever the heat balance exceeds indicated neutron power by more than 2%. In effect, this change request limits the plant to 2% steady-state neutron power error, with margin to a total 4% error immediately following maneuvering transients.

Question No. 5

Provide an updated power map which reflects the additional cycle 3 burnup and the non-cross-core shuffle for cycle 4.

Response:

See attached Figure 5-1.

FIGURE 5-1. BOC (4 EFPD), CYCLE 4 TWO-DIMENSIONAL RELATIVE POWER DISTRIBUTION-FULL POWER, EQUILIBRIUM XENON, APSRs INSERTED

POOR ORIGINAL

	8	9	10	11	12	13	14	15
H	0.94	1.05	1.23	0.97	1.21	0.91	0.83	0.79
K		1.24	1.06	1.16	1.05	1.18	0.85	0.79
L			1.32	1.05	8 1.05	0.93	1.18	0.69
M				0.99	1.16	0.95	0.99	
N					1.03	1.13	0.70	
O						0.55		
P								
R								

Updated power map reflecting additional Cycle 3 burnup (287.1 EFPD) and non-cross-core shuffle.

X	INSERTED ROD GROUP NUMBER
X.XX	RELATIVE POWER DENSITY

1489 017

Question No. 6

Provide or reference the bounding transient and accident analyses during bleed and feed operation.

Response:

The parameters having the greatest effect on the Safety Analysis are the core-thermal parameters, thermal hydraulic parameters, and kinectic (including feedback coefficients) parameters. As shown in Tables 4-2, 6-1, and 7-1 and discussed in Section 7 of the Reload Report, the FSAR Safety Analysis is still a bounding analysis. The one exception is a slightly less initial boron concentration and the effects of that are discussed in the response to Question 3.

1489 018

Question No. 7:

Provide an explanation of the increase in quadrant tilt from 3.41 to 4.92% being proposed in the Technical Specifications. What kind of a penalty is taken in the calculation of peaking factors in order to account for the allowable 4.92% tilt? Provide the basis for the adequacy of this penalty.

Response:

As indicated in Item 2 of Section 8 of the TMI-1, cycle 4 Reload Report, the quadrant tilt limit for cycle 4 was returned to the original limit value of 4.92% actual core tilt used in cycles 1 and 2. The reason for the tighter limit, 3.41% in cycle 3 was that in order to preserve flexible operating windows for imbalance and control rod position, a smaller peaking penalty (5.1%) for allowable quadrant tilt was used to offset the required peaking penalty due to potential fuel rod bow. Thus, the allowable tilt limit was correspondingly reduced. The TMI-1, Cycle 3 Reload Report discussed these items in Section 8.

For Cycle 4, a trade-off of this type was not necessary due to the use of a statistical combination of peaking factors (Section 8, Item 3), the removal of the densification power spike from consideration in setting ECCS-dependent Technical Specification limits (Section 8, Item 4), and the reduced peaking behavior of the Cycle 4 core design. Thus, the original 4.92% limit on quadrant tilt and its associated peaking penalty (7.36% or a peaking factor of 1.0736) were reinstated.

The peaking factors quoted in the preceding paragraphs were derived from the relationship established for the increase in the peak power due to a quadrant power tilt. The following discussion describes the calculations which have been performed to investigate this behavior.

The data on calculated power peak increase due to quadrant power tilt are presented in the attached figure. These data are from both Figure 3-5 of BAW-10078 and recent investigations of the Oconee I, Cycle 4 tilt behavior. The following discussion characterizes the method of tilt inducement used in the various calculations.

The calculations were performed in both 2-D and 3-D full core geometry using the PDQ07 and FLAME 3 computer codes. Two dimensional geometry was used whenever the tilt effects were uniform axially. In these cases the radial peak change conservatively reflected the total peak change. This fact was confirmed by selected 3-D check cases. The value of tilt against which the peak increase was plotted was obtained by integrating the mesh block or nodal powers to get the power produced in each quadrant. The expression for tilt is:

$$\% \text{ Quadrant Tilt} = \left(\frac{\text{Quadrant Power}}{\text{Average Quadrant Power}}\right) - 1 \times 100,$$

and for the attached figure represents what can be called the "actual" quadrant tilt.

1489 019

Response to Question No. 7 continued:

Following the legend in the attached figure, the first tilt type considered was that due to multiple rods out of sequence (symbol x). Two of these values are from Figure 3-5 of BAW-10078, and one from recent 3-D FLAME investigations of potential Oconee I multiple misaligned rods. These three [illegible] represent from 2-6 rods misaligned. In the Oconee I case, rods in diagonally opposite quadrants were moved in opposite directions. The core was modeled with 24 axial nodes of 6" each. Bank 7 was misaligned such that one rod (on a minor axis) was one node above the bank average and the diagonally opposite rod was one node below the bank average.

The next type of tilt, shown with the symbol △, was that caused by a dropped rod. In addition to the four cases from Figure 3-5 of BAW-10078, eleven additional cases were calculated for the Oconee I, Cycle 4. Every potential dropped rod location, including those on the major axes, was investigated.

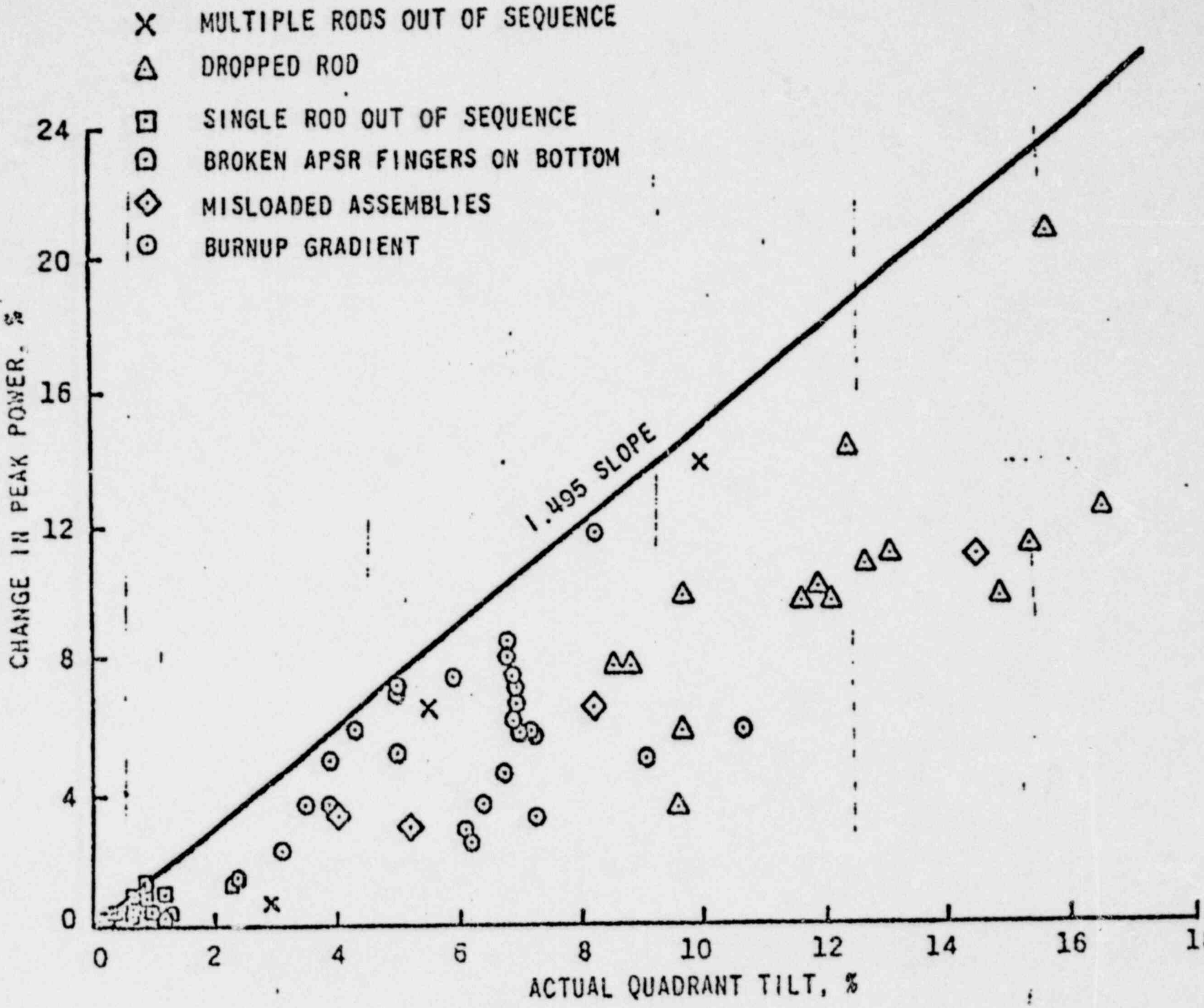
The third tilt type was that caused by a single rod out of sequence (symbol ⊡). These ten cases were a[illegible] reported in BAW-10078. The results are all clustered at low tilt and peak increase values. These were 3-D PDQ07 cases.

The fourth tilt type shown (symbol ⌂) was that due to various numbers of individual APSR fingers (1-3) assumed to be broken off and resting on the bottom in three different assembly locations. Three-dimensional FLAME calculations for the beginning of Oconee I, Cycle 4 were run at 40% FP, and without xenon, to amplify peaking effects.

The fifth tilt type was generated assuming several (3-6) misloaded assemblies (symbol ◇). Enrichment deviations of from ± .01 w/o (6 locations) to -.90 w/o (3 locations) were investigated. Again, the beginning of Cycle 4 of Oconee I was the configuration analyzed.

The sixth and final tilt type investigated (symbol ⊙) was that caused by a non-symmetric burnup distribution in two fuel batches being carried over into Cycle 4 of Oconee I. Partial results of these calculations are given in BAW-1477. FLAME was used to simulate an end of Cycle 3 burnup asymmetry of +2% in one core quadrant and -2% in the diagonally opposite quadrant. The fuel was then shuffled into the Cycle 4 pattern and depleted in full core geometry to 50 EFPD. The power level was set at 40% FP to 4 EFPD, at 75% FP from 4 to 23 EFPD, and a 100% FP from 23 to 50 EFPD. A total of 26 variations of power level and burnup supplied data for the points plotted.

As can be observed from the figure, all of the over 60 data points fall below the line which has a slope of 1.495. This was the value assumed in assigning a 7.36% peak increase to an allowable tilt of 4.92% for the TMI-1, Cycle 4 Technical Specification.



Response to Question No. 7

1489 021

Question No. 8

How many orifice rod assemblies will be present during cycle 4? Where will they be located? What are the peaking factors and flow problems associated with removal of orifice rod assemblies?

Response:

There will be 62 orifice rod assemblies present during cycle 4.

Orifice rod (OXX) locations are indicated on the attached TMI-1 Cycle 4 Core Loading Plan.

The removal of orifice rod assemblies does not affect core peaking distributions; furthermore, no orifice rod assemblies have been removed relative to previous reload cycles. The absence of 44 orifice rod assemblies has been factored into core thermal-hydraulic analyses by a reduction in the reactor coolant flow available for heat transfer. The core thermal hydraulic analyses presented in the TMI-1 FSAR (Reload Report, reference 1) and Fuel Densification Report (Reload Report, reference 4) were based upon a maximum core bypass flow of 6.04% of system flow. The current thermal-hydraulic analysis basis, as used for licensing of cycles 2, 3, and 4, includes a core bypass flow of 8.34% of system flow, with the additional 2.3% bypass being a result of the absence of 44 ORA's. The actual core coolant flow available for heat transfer is greater than that which had been assumed for FSAR analyses by virtue of the fact that the RCS flowrate is approximately 109% of design flow. This is reflected in part by the use of 106.5% of design flow as the basis for thermal-hydraulic analyses of cycles 2, 3, and 4.

Response to Question No. 8

POOR ORIGINAL

TMI-1 Cycle 4
Core Loading Plan
Rev. 2

FUEL TRANSFER CANAL →

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						OLK	140	13W	OLZ	14P					
B				145 OXX	13Z	OMI C58	OAG	OAH C16	OAX	13R C14	13V	141 R00005			
C			OBL	OMS OXX	OAU C29	1B43 OXX	OLG C09	1A22 OXX	OLV C17	1B29 OXX	OBA C45	13F OXX	OBG		
D		134	13C OXX	OBJ C55	OLN OXX	OLA A04	OAM OXX	OL6 C43	OBC OXX	OMH A05	OLK OXX	OAN C57	OLR OXX	142	
E		137	OAB C41	OLP OXX	1A35 C30	OAQ OXX	OMF C44	1A08 OXX	OME C46	OAJ OXX	1A34 C33	OMU OXX	OAK C32	13X	
F	OMW	13S C08	1B31 OXX	OM9 A03	OAT OXX	OLT C13	OAY OXX	OM7 C31	OA5 OXX	OLH C19	OBD OXX	OLF A06	1B30 OXX	OMQ C60	OLC
G	13K	OBM	OLY C11	OB2 OXX	OMR C42	OA8 OXX	OMA C12	OB6 OXX	OM2 C18	O3S OXX	OMH C48	OAS OXX	OLU C15	OA9	OMM
H	13Y	OBE C10	1A09 OXX	OM5 C39	1A13 OXX	OMO C28	CAA OXX	1A11 C61	OBR OXX	OL9 C34	1A02 OXX	OM4 C47	1A50 OXX	OB5 C22	13J
K	OMV	OA7	OLX C03	OAD OXX	OMJ C40	OBO OXX	OM1 C06	OB3 OXX	OMS C24	OBN OXX	OMP C50	OAC OXX	OM6 C23	OAW	13E
L	[illegible]	OMG C56	1B25 OXX	OMB A02	OB4 OXX	OLM C07	OAP OXX	OL7 C25	OB9 OXX	OMF C01	OBH OXX	OLD A07	1B27 OXX	14Q C20	144
M		138	OB8 C26	OLJ OXX	1A18 C27	OAC OXX	OML C38	1A32 OXX	OMK C52	OBQ OXX	1A14 C36	OMC OXX	OBP C49	139	
N		143	13A OXX	OAZ C53	OLQ OXX	OMY A01	OBK OXX	OM3 C51	OAL OXX	OMJ A08	OMD OXX	OAV C59	13U OXX	13D	
O			OAE	13B OXX	OBB C37	1B36 OXX	OLW C05	1A21 OXX	OLS C21	1B33 OXX	OAR C35	OLL OXX	OB1		
P				OMX R00006	136	OMI C02	OAF	OBF C04	OL7	13T C54	13M	OL8 OXX			
R						146	135	13G	13H	OLB					

Z

Fuel I.D. (1AXX-Batch 1; 1BXX-Batch 2; OAX or OBX-Batch 4; Others-Batches 5 & 6)
Orifice (OXX) or Control Rod I.D. (CXX - CRA; AXX - APSRA)
Source I.D. (R000XX - Secondary)

Note: NJO Prefixes Batch 4, 5, and 6 F.A. I.D.'s.

Question No. 9:

What is the maximum impact energy (in ft-lb) corresponding to the alarm setpoints currently used in the Loose Parts Monitoring System? Also, briefly describe the location of the accelerometers.

Response:

The nominal impact energy corresponding to the alarm setpoints currently used in the Loose Parts Monitoring System is 0.5 ft lb. The location of the accelerometers is as follows:

a) Lower reactor vessel-incore guide tube 5

b) Lower reactor vessel-incore guide tube 13

c) Upper reactor vessel-reactor vessel head shroud

d) Upper reactor vessel-reactor vessel head shroud

e) Steam generator "A"-upper tube sheet north side

f) Steam generator "B"-upper tube sheet south side

g) Steam generator "A"-upper tube sheet south side

h) Steam generator "B"-upper tube sheet north side

Question No. 10:

Provide the following information regarding measurements made during cycle 3.

a) Provide a low and high power XY power map for BOC 3. Both measured and predicted assembly powers should be given.

b) Provide the measured and predicted BOC 3 rod bank worths by bank.

c) Provide the BOC 3 measured values for critical boron concentration and moderation temperature coefficient. State the power and xenon conditions under which each measurement was taken.

d) Provide the measured and predicted ejected rod worth for BOC 3. State the condition under which the test was done.

Response:

a) Power distributions from BOC-3 physics testing are provided on Figures 1, 2, 3 and 4 attached. Figures 1 and 2 provide radial and total peaking factors at 41.8% full power. Figures 3 and 4 provide radial and total peaking factors at 99.24% FP.

b) The measured and predicted rod bank worths from BOC-3 zero power physics tests are as follows:

	Predicted	Measured
Group 7	0.73% ΔK/K	0.76% ΔK/K
Group 6	0.96% ΔK/K	1.01% ΔK/K
Group 5	1.08% ΔK/K	1.13% ΔK/K
Group 1 - 4	5.82% ΔK/K	5.48% ΔK/K

c) The critical boron concentration at BOC-3 was measured at zero power and xenon free conditions. The All Rods Out Concentrations are as follows:

	Predicted	Measured
ARO Boron	1280 ppm	1249 ppm

The results of the three moderator coefficient tests performed during BOC-3 testing are as follows:

	Predicted	Measured
α m (zero power, Xenon free, 1255 ppmB)	$<5\times10^{-3}$% ΔK/K/°F	$+2.$[illegible]5×10^{-3}% ΔK/K/°F
α m (zero power, Xenon free, 1005 ppmB)	-4.7×10^{-3}% ΔK/K/°F	-4.8×10^{-3}% ΔK/K/°F
α m (75% FP, 3-D equilibrium Xenon, 818 ppmB)	-11×10^{-3}% ΔK/K°F	-9×10^{-3}% ΔK/K°F

d) The zero power maximum ejected rod worth measurement was made at BOC-3 zero power, no Xenon, 532°F Tave with Control Rod Groups 5, 6 and 7 at 0% withdrawn. The results are as follows:

	Predicted	Measured
Ejected Rod Worth	0.34% ΔK/K	0.34% ΔK/K

FIGURE 1

RADIAL PEAKING FACTORS

	8/H	9/G	10/F	11/E	12/D	13/C	14/B	15/A
H/8	1.013 1.04	1.208 1.21	0.967 1.02	1.12 1.20	1.289 1.20	1.219 1.16	0.608 0.57	0.757 0.77
K/9		0.974 1.03	0.959 0.99	1.008 1.08	1.377 1.39	1.25 1.22	1.088 1.10	0.916 0.86
L/10			0.982 0.97	1.090 1.06	1.192 1.18	0.9[illegible] 0.95	1.389 1.32	0.821 0.73
M/11				1.214 1.28	1.102 1.16	0.939 0.94	0.971 1.01	
N/12					0.59 0.57	0.687 0.66	0.547 0.61	
O/13						0.485 0.49		

LEGEND

X.XXX	Measured Value
X.XX	Calculated Value

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Maximum Measured Value 1.389 (L-14)

Maximum Calculated Value 1.39 (K-12)

Maximum Error (%):

$\frac{1.389-1.39}{1.39} \times 100 = -0.07\%$

	% wd.
Gp. 1-4	100
Gp. 5	100
Gp. 6	86
Gp. 7	9
Gp. 8	31

Power Level 41.8 %FP

Effective Full Power Days 0.56 EFPD

Figure 2

RADIAL PEAKING FACTORS

	8/H	9/G	10/F	11/E	12/D	13/C	14/B	15/A
H/8	1.292 1.31	1.49 1.47	1.191 1.25	1.316 1.42	1.502 1.45	1.384 1.35	0.683 0.66	0.872 0.92
K/9		1.17 1.27	1.131 1.25	1.381 1.28	1.611 1.67	1.459 1.44	1.243 1.29	1.094 1.04
L/10			1.15 1.18	1.278 1.29	1.525 1.60	1.114 1.14	1.641 1.58	0.988 0.88
M/11				1.452 1.54	1.301 1.41	1.056 1.10	1.132 1.20	
N/12					0.723 0.72	0.766 0.76	0.651 0.71	
O/13						0.567 0.57		

LEGEND

X.XXX	Measured Value
X.XX	Calculated Value

Maximum Measured Value 1.641 (L-14)

Maximum Calculated Value 1.67 (K-12)

Maximum Error (%):

$\frac{(1.641-1.67)}{1.67} \times 100 = -1.74\%$

	% wd.
Gp. 1-4	100
Gp. 5	100
Gp. 6	86
Gp. 7	9
Gp. 8	31

Power Level 41.8 %FP

Effective Full Power Days 0.56 EFPD

Figure 3

RADIAL PEAKING FACTORS

	8/H	9/G	10/F	11/E	12/D	13/C	14/B	15/A
H/8	1.028 1.03	1.215 1.19	1.026 1.00	1.123 1.18	1.285 1.18	1.22 1.15	0.624 0.58	0.772 0.79
K/9		0.985 1.02	0.975 0.98	0.961 1.06	1.368 1.36	1.246 1.21	1.082 1.10	0.905 0.89
L/10			0.991 0.96	1.097 1.05	1.187 1.17	0.962 0.97	1.351 1.32	0.811 0.75
M/11				1.206 1.26	1.111 1.15	0.938 0.95	0.966 1.03	
N/12					0.609 0.58	0.70 0.68	0.559 0.63	
O/13						0.495 0.52		

LEGEND

X.XXX	Measured Value
X.XX	Calculated Value

POOR ORIGINAL

Maximum Measured Value 1.368

Maximum Calculated Value 1.36

Maximum Error (%):

$$\frac{1.368-1.36}{1.36} \times 100 = +0.59\%$$

	% wd.
Gp. 1-4	100
Gp. 5	100
Gp. 6	88
Gp. 7	13
Gp. 8	20

Power Level 99.24 %FP

Effective Full Power Days 5.42 EFPD

Imbalance - 2.28

Response to Question No. 10

Figure 4

RADIAL PEAKING FACTORS

	8/H	9/G	10/F	11/E	12/D	13/C	14/B	15/A
H/8	1.297 1.30	1.49 1.47	1.19 1.26	1.286 1.44	1.498 1.48	1.36 1.37	0.771 0.75	0.881 0.95
K/9		1.163 1.27	1.113 1.27	1.31 1.31	1.632 1.70	1.431 1.45	1.281 1.29	1.048 1.06
L/10			1.131 1.20	1.276 1.32	1.456 1.64	1.112 1.18	1.591 1.60	0.959 0.91
M/11				1.391 1.55	1.278 1.43	1.056 1.13	1.125 1.22	
N/12					0.842 0.81	0.769 0.79	0.664 0.74	
O/13						0.556 0.61		

LEGEND

X.XXX	Measured Value
X.XX	Calculated Value

Maximum Measured Value 1.632

Maximum Calculated Value 1.70

Maximum Error (%):

$\frac{1.632-1.70}{1.70} \times 100 = -4.0\%$

	% wd.
Gp. 1-4	100
Gp. 5	100
Gp. 6	88
Gp. 7	13
Gp. 8	20

Power Level 99.24 %FP

Effective Full Power Days 5.42 EFPD

Imbalance - 2.28

1489 029

Question No. 11:

The startup physics test program as given in Section 9 lacks the necessary depth of discussion. A significant amount of additional detail will be required in order to make clear the acceptability of the methods, procedures and acceptance criteria used for the various tests. Specifically, the following questions are submitted on the test programs.

Response:

The methods, detailed procedures and acceptance criteria for the BOC Physics Testing Program at TMI-1 have been reviewed in detail by the NRC Region I Office of Inspection and Enforcement staff throughout Cycle 1 (initial start-up-and mid cycle rod swap program) Cycle 2 and Cycle 3. The methods and procedures used for physics testing and adherance to acceptance criteria have been noted to be acceptable. These methods have not changed for BOC-4 Physics Testing. Controlled copies of the detailed procedures describing the methods and acceptance criteria for each test used for BOC physics testing are available on site for your review along with all data analysed to date. Summaries of the test method and acceptance criteria for each of the tests identified in your enclosure are as follows:

Question No. 11(a):

Describe in detail the tests being done to check for a misloaded assembly. What assurances are there that the core is as expected before going to powers > 5% rated power?

Response:

After completion of the fuel shuffle, prior to installation of the reactor vessel head, a video map is made of each fuel assembly identification. This video map is then compared to the Cycle loading plan to assure that each fuel assembly is in its designated core position.

Question No. 11(b):

Describe the procedures for the control rod-trip test. Include the acceptance criteria and the procedures to be followed if the acceptance criteria are not met.

Response:

The control rod trip times are measured in accordance with Technical Specification 4.7.1 by measuring the time from deenergizing the undervoltage trip device until the 25% withdrawn (3/4 insertion) reed switch is actuated.

The acceptance criteria is 1.66 seconds for hot full flow or 1.40 seconds for hot no flow conditions.

If the acceptance criteria is not met for a specific rod, the rod is declared inoperable until the problem is resolved.

1489 030

Question No. 11(c):

Provide the details of the procedures for the critical boron concentration tests. Discuss how corrections are made to the measured data and how the measured data is compared to the predictions. What are the acceptance criteria and what are the procedures if the acceptance criteria are not met?

Response:

Initial criticality following a fuel reload is achieved by withdrawal of control rods in Group 1-6 to 100% and Group 7 to 75%, followed by deboration of the reactor coolant. Once an equilibrium boron sample is obtained at the initial critical rod position (normally 75% withdrawn on Group 7) the All Rods Out Critical Boron Concentration is obtained by fully withdrawing Group 7 control rods and measuring the doubling times due to the reactivity addition. This reactivity is converted to an equivalent boron concentration and is added to the equilibrium boron concentration obtained at initial criticality to obtain an actual-all rods out equivalent boron concentration. The predicted results for BOC-4 start-up is 1250 ppm. If the acceptance criteria for this test ($\pm$ 100 ppm) is exceeded the reactor would be placed in hot shutdown (Keff < 0.99) and the results would be evaluated in depth prior to regaining criticality.

Question No. 11(d):

Describe in detail the procedures and methods used for the temperature reactivity coefficient tests. Also provide the acceptance criteria and the procedures to be followed if the acceptance criteria are not met.

Response:

The temperature coefficient of reactivity is measured during BOC Zero Power Physics testing at two boron concentrations (All Rods Out and at the Minimum Rod Insertion Index). With the reactor just critical at equilibrium reactor coolant system conditions, the reactor coolant system average temperature (Tave) is varied $\pm 5^{\circ}F$. The change in net core reactivity due to the variation in Tave is measured by the Reactimeter (a reactivity calculator which uses input from an intermediate neutron range detector). The control rods are not moved during this test at zero power. The reactivity change per change in $^{\circ}F$ is calculated and extrapolated to 100% full power. If the extrapolated value shows that the moderator coefficient would be positive at hot full power, the temperature coefficient test will be repeated at 75% full power and again extrapolated. If the extrapolation reveals the moderator coefficient will be negative at hot full power, the temperature coefficient test is repeated at 100% full power to verify that the acceptance criteria has been met. Temperature coefficient measurements at power are performed by varying Tave and observing the change in control rod position while maintaining constant reactor power. Thus the change in reactivity based on differential rod worth per change in reactor coolant system average temperature is calculated. The predicted result of the BOC Zero power isothermal temperature coefficient is $- 5.3 \times 10^{-4}$ % $\Delta K/K/^{\circ}F$ at 1230 ppm boron. The moderator coefficient shall be less than $+0.5 \times 10^{-4}$ % $\Delta K/K/^{\circ}F$ at zero power to assure a non-positive moderator coefficient above 95% full power. Results of each test would be evaluated if acceptance criteria were not met and reactor power would not be increased above 95% full power until it could be shown that a non-positive moderator coefficient existed.

1489 031

POOR ORIGINAL

Question No. 11(e):

Provide the details of the regulating control rod group reactivity worth tests. Give the predicted worth of each group to be measured, and the stuck rod worth and the predicted total worth for all rods. Also provide the acceptance criteria and the procedures to be followed if the acceptance criteria are not met.

Response:

Control rod group reactivity measurements are performed at hot zero power conditions using the boron/rod swap method and the rod drop method. The boron/rod swap method is used to measure the differential and integral reactivity worths of control rod groups 5, 6 and 7. The total reactivity worth of the safety rod groups (Groups 1-4) is measured by the rod drop method. The boron/rod swap method consists of establishing a deboration rate in the reactor coolant system and compensating for the reactivity changes of this deboration by inserting control rod groups 7, 6 and 5 in incremental steps. The reactivity changes that occur during these measurements are calculated based on reactimeter data and differential rod worths are obtained from the known reactivity worth versus the change in rod group position. The differential rod worth of each of the controlling groups are then summed to obtain integral rod group worths. For the rod drop measurement of the worth of Groups 1-4, critical equilibrium conditions are established with control rod groups 1-4 withdrawn from the core to the minimum rod index. The control rod groups being measured are then dropped into the core. The reactivity inserted into the core is then calculated by analyzing data obtained from the reactimeter. The total reactivity worth of groups 1-4 is measured using the rod drop method.

The predicted group worths for BOC-4 testing are as follows:

Group 7	1.37% ΔK/K
Group 6	0.95% ΔK/K
Group 5	1.39% ΔK/K
Group 1-4	5.00% ΔK/K

The worst case predicted stuck rod worth Cycle 4 is 2.03% ΔK/K.

The acceptance criteria for total worth is ± 10% for Groups 5-7 and ± 15% for Groups 1-4. The total rod worth derived from these measurements is used to determine available shutdown margin. Shutdown margin must be greater than 1% ΔK/K considering the most reactive stuck rod out of the core.

Question No. 11(f):

Describe in detail the procedures for the ejected control rod reactivity worth test. State the methods used to compare the measurements with predicitions and the acceptance criteria. Also, include procedures if the acceptance criteria are not met.

Response:

Ejected control rod worth is measured at hot zero power conditions with the controlling rod groups at the minimum allowable rod index using two techniques.

Response to Question No. 11(f) continued:

The first technique is the boron swap method during which the boron concentration of the reactor coolant system is slowly and continuously increased. The ejected rod is withdrawn in quick steps to compensate for the reactivity inserted by the boration and the reactivity change is measured by a reactivity calculator. The sum of the incremental reactivity changes gives the total worth of the ejected rod. In the second technique (rod swap method), critical equilibrium conditions are established with the ejected rod withdrawn to 100%. The ejected rod is then inserted into the core by swapping reactivity with another rod group. The measured instantaneous worth of the rod (using reactivity calculator) is taken as the worth of the ejected rod.

These measured values are then error adjusted for uncertainty associated with the use of predicted rod worth data and uncertainty associated with the use of the boron swap method. This error adjusted maximum ejected rod worth is then compared with acceptance criteria.

If the acceptance criteria of this procedure was not satisfied, the reactor would be taken to hot shutdown condition and the results would be evaluated.

Question No. 11(g):

TMI-1 had a quadrant tilt at the beginning of Cycle 3. How did this tilt change during the cycle? How was the presence of this tilt used in the predictions of the power distribution for Cycle 4?

Response:

The TMI-1 Cycle 3 indicated tilt remained below the error-adjusted Technical Specification limit throughout the entire cycle. The indicated tilt at the beginning of the cycle was near 1% and it remained steady for approximately 120 EFPD's. After returning to power following an outage, the indicated tilt increased to 2.2%. It gradually decreased to 1.2% by the end of the cycle. Because of the apparent enhancement of tilt in another plant due to cross core shuffling the original cycle 4 design was revised to a shuffle philosophy which generally moves the fuel from a given quadrant into both of the adjacent quadrants. This shuffle tends to minimize possible carry-over effects of any burnup assymmetry that might be present in the previous cycles. Because of the low value of indicated tilt at end of cycle 3, any carry-over effects of the tilt would be small and should be essentially eliminated by the revised fuel shuffle.

Consequently, the presence of tilt in cycle 3 was not used in the prediction of the power distribution for Cycle 4.

POOR ORIGINAL

Question No. 11(h):

Provide the details of the core power distribution tests. Describe in detail the methods used to predict the assembly by assembly power as well as the analyses of the data obtained during the measurements. What are the assembly by assembly acceptance criteria? How are tilts accounted for in the analysis of the data? If a 1/4 or 1/8 core map is the result of the measurement, what method is used to determine the assembly power for those assemblies having their symmetric assemblies instrumented? For example, are the measured assembly powers averaged, or is only one of the symmetric measurements used?

Response:

Core Power Distribution Tests are performed at 40, 75, and 100% FP. The test at 40% FP is essentially a check on power distribution in the core to bring attention to any abnormalities before escalating to the 75% FP plateau. Rod index is established at a nominal full power configuration which is where the core power distribution calculations are performed. APSR position is established to provide a core power imbalance corresponding to the imbalance where the core power distribution calculations are performed.

The following acceptance criteria are placed on the 40% FP test:

(1) The worst case maximum linear heat rate must be less than the LOCA limit specified in Technical Specifications Figure 3.5-2J.

(2) The minimum DNBR must be greater than 1.30.

(3) The value obtained from the extrapolation of the minimum DNBR to the next power plateau overpower trip setpoint must be greater than 1.30 or fall outside the RPS power/imbalance trip envelope.

(4) The value obtained from the extrapolation of the worst case maximum linear heat rate to the next power plateau overpower trip setpoint must be less than the fuel melt limit or fall outside the RPS power/imbalance trip envelope.

(5) The quadrant power tilt shall not exceed the limits specified in Technical Specifications.

(6) The highest measured radial peak shall not exceed the highest predicted radial peak by more than 8%.

(7) The highest measured total peak shall not exceed the highest predicted total peak by more than 12%.

Items 1, 2, 5, 6, and 7 above are established for the purpose of verifying core nuclear and thermal calculational models, thereby verifying the acceptability of data from these models for input to safety evaluations.

Items 3 and 4 establish the criteria whereby escalation to the next power plateau may be accomplished without exceeding any safety limits specified by the safety analysis with regard to DNBR and linear heat rate.

Response to Question 11(h) Continued

The tests are also performed at 75 and 100% FP and in the same manner as the 40% FP test with one exception. At 75 and 100% FP, three dimensional xenon equilibrium is required; whereas, at 40% FP there are no equilibrium xenon requirements. The same acceptance criteria apply with the exception that the highest measured radial and total peaks shall not exceed the highest predicted radial and total peaks by more than 5 and 7.5%, respectively, for both 75 and 100% FP testing. The more restrictive limits are due to the equilibrium xenon requirements at 75 and 100% FP.

Predictions for the radial and total peaks at 40, 75, and 100% FP are calculated using the FLAME-3 with thermal-hydraulic feedback code (BAW-10124). Radial peaks are calculated from the predicted power output for each assembly in a 1/8 core. Total peaks are calculated from the predicted power output of the maximum segment for each assembly in a 1/8 core.

Assembly and segment power representations are calculated by the on-line computer based on current-signal outputs from the 52 incore detector strings. Any tilt which exists in the core is inherent in the measurement of neutron flux by the incore detector system. Only instrumented assemblies are utilized in the analysis of the data to calculate measured radial and total peaks for comparison to predicted radial and total peaks. Symmetric instrumented locations are averaged to provide a single value for the assembly or segment power in the 1/8 core location. Radial and total peak are then calculated. As previously stated, the maximum measured radial and total peaks are compared to maximum predicted radial and total peaks. There are no criteria for comparisons on an assembly by assembly basis.

Tilt effects are accounted for in the calculation of DNBR and linear heat rate. If a tilt does exist, a routine in the on-line computer adjusts the segment power representations of an instrumented assembly in order to provide segment power representations of a symmetric, non-instrumented assembly. DNBR and linear heat rate are calculated by the on-line computer for the maximum assembly in each of the four core flow regions. These values are then compared to acceptance criteria previously discussed. In addition, a hand calculation of linear heat rate is performed in order to obtain values for comparison with LOCA acceptance criteria which are level dependent.

Question No. 11(i):

Provide a commitment to prepare a brief summary report of the Cycle 4 physics startup tests and to submit this report to NRC within 45 days of the completion of the startup tests. This report should include both measured and predicted values. If the difference between the measured and predicted values exceed the acceptance criterion, the report should discuss the adequacy of the actions taken.

Response:

Met-Ed will provide the information requested above for the Cycle 4 physics startup tests, and will submit the information to NRC within 90 days of the completion of the startup tests, consistant with other Tech Spec reporting requirements.