P. 04/14/n8 ●

REC: REID R '   ORG HERBEIN J G   DOCDATE: 04/10/79
  NRC   METROPOL EDISON   DATE RCVD: 04/13/79

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SUBJECT: LTR 1   ENCL 1
RESPONSE TO NRC LTR DTD 04/07/79...FORWARDING ANSWERS TO THE CYCLE 4 RELOAD
QUESTIONS AND ADDL CONCERNS AS REQUESTED BY NRC.

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1489 009

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April 10, 1978
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TELEPHONE 215 – 929-3601

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

781030058

1489 010

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 flowrate 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.

TABLE 1

INTERVAL BETWEEN PUSH-PULL LOAD CHANGES

Interval Days	Load Change Percentage	Last FPD Conc. (PPM)	Last Time In Life		Fluid to be Processed Gallons
			FPD	%	
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	

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	1.637	(BOL)	8.2%	(BOL)
(January 9, 1978)	1.421	(EOL)	20.3%	(EOL)
Non-Cross Core Shuffle	1.596	(BOL)	10.5%	(BOL)
	1.407	(EOL)	21.1%	(EOL)
Non-Cross Core Shuffle and	1.547	(BOL)	13.2%	(BOL)
extended Cycle 3 (287.1 EFPD) (To be included in revised Reload Report)	1.403	(EOL)	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.

Response to Question No. 2

Figure 2-1. 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		1.06 /8	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								

Inserted Rod Group Number

Relative Power Density

1489 014

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.
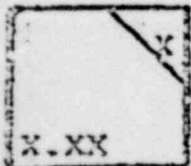
Response:

See attached Figure 5-1.

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

	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	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

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.

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 the e 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 ach 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.

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
~ 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 ' reported in BAW-10078. The results are all clustered
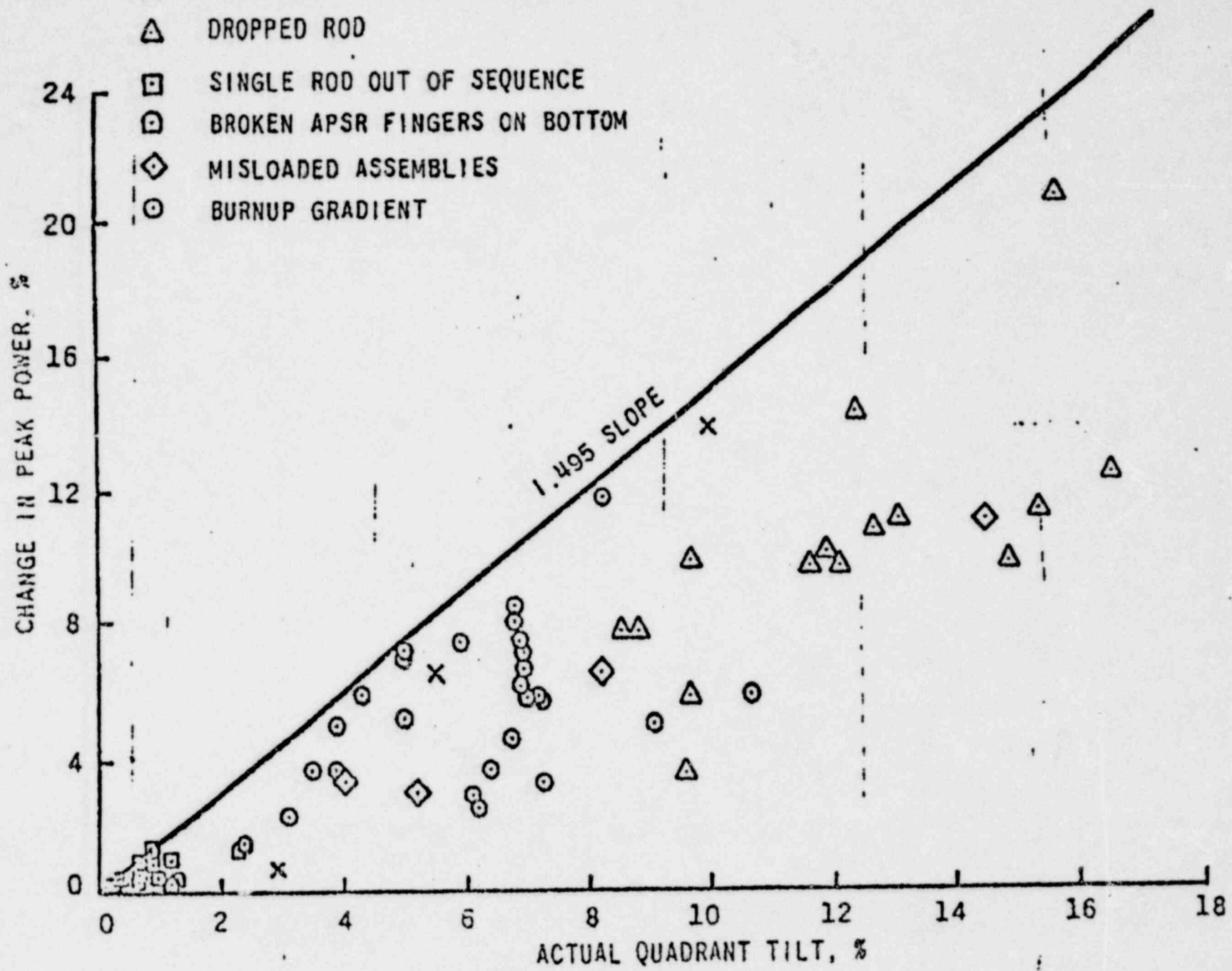at low tilt and peak increase values. These were 3-D PDQ07 cases.

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.

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.

TMI-1 Cycle 4
Core Loading Plan
Rev. 2

FUEL TRANSFER CANAL ————————▷

X

Row	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						OLE	140	13W	OLZ	14P					
B				145 OXX	13Z	OXI C58	OAG	OAH C16	OAX	13R C14	13V	141 R00002			
C			OBL	OXS C29	OAU	1B43 C09	OLG	1A22	OLV C17	1B29	OBA C45	13F OXX	OBG		
D		134	13C C55	OBJ OXX	OLN	OIA A04	OAM	OLG C43	OBC	OWH A05	OLK	OAN C57	OLR OXX	142	
E		137 C41	OAB	OLP C30	1A35	OAQ	OMF C44	1A03	OME C46	OAJ	1A34 C33	OAU	OAK C32	13X	
F	OMW	13S C08	1B31 OXX	OM9 A03	OAT	OLT C13	OAY	OM7 C31	OA5	OLH C19	OBD	OLF A06	1B30 OXX	OMQ C60	OLC
G	13K	OBM C11	OLY	OB2 C42	OMR	OA8 C12	OMA	OB6 C18	OM2	OS3 C48	OMH	OAS C15	OLU	OA9	OXX
H	13Y	OBE C10	1A09 OXX	OX5 C39	1A13	OMO C28	OAA	1A11 C61	OBR	OL9 C34	1A02 OXX	OM4 C47	1A50 OXX	OB5 C22	13J
K	OMV	OA7 C03	OLX OXX	OAD C40	OMJ	OBO C06	OM1	OB3 C24	OMS	OBN C01	OMP C50	OAG OXX	OM6 C23	OAW	13T
L	OMG	OMG C56	1B25 A02	OMB OXX	OB4	OLM C07	OAP	OL7 C25	O39	OMT C01	OBH OXX	OLD A07	1B27 OXX	140 C20	144
M		138 C26	OB3	OLJ OXX	1A18 C27	OAC OXX	OML C33	1A32 OXX	OXX 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	OWJ AOS	OMD OXX	OAV C59	13U OXX	13D	
O			OAE	13E OXX	OBB C37	1B36 OXX	OLW C05	1A21 OXX	OLS C21	1S33 OXX	OAR C35	OLL OXX	OB1		
P				OXX R00008	136	OXI C02	OAF	OBF C04	OL7	13T C54	13M	OL6 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)	$<5 \times 10^{-3}$% ΔK/K/°F	$+2.25 \times 10^{-3}$% 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?? / 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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% wd.

Maximum Measured Value 1.389 (L-14)   Gp. 1-4   100

Maximum Calculated Value 1.39 (K-12)   Gp. 5   100

Maximum Error (%):   Gp. 6   86

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

POOR ORIGINAL

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 41.3 %FP

Effective Full Power Days 0.56 EFPD

1489 027

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

1489 028

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 |

POOR ORIGINAL

% wd.

Maximum Measured Value __1.632__   Gp. 1-4 __100__

Maximum Calculated Value __1.70__   Gp. 5 __100__

Maximum Error (%):   Gp. 6 __88__

$\dfrac{1.632-1.70}{1.70} \times 100 = -4.0\%$ Gp. 7 __13__

Gp. 8 __20__

Power Level __99.24__ %FP

Effective Full Power Days __5.42__ EFPD

Imbalance - 2.26

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.

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 re-
activity 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 $-5^{+}\,^{\circ}F$. 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 re-
peated 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 con-
stant reactor power. Thus the change in reactivity based on differential
rod worth per change in reactor coolant system average temperature is calcu-
lated. 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 coef-
ficient 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.

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 re-
activity 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 dif-
ferential 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% $\Delta K/K$
Group 6	0.95% $\Delta K/K$
Group 5	1.39% $\Delta K/K$
Group 1-4	5.00% $\Delta K/K$

The worst case predicted stuck rod worth Cycle 4 is 2.03% $\Delta K/K$.

The acceptance criteria for total worth is $\pm$ 10% for Groups 5-7 and $\pm$ 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% $\Delta K/K$ 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 assymetry 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.

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 instrumated 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.