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March 1, 1979
GQL 0298

Director of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. R. W. Reid, Chief
Operating Reactors Branch No. 4

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Cycle 5 Reload - Additional Information

Your letter of February 16, 1979, requested additional information to support the TMI-1 Cycle 5 Reload. Enclosed, please find Met-Ed's responses to your requests.

Draft responses were transmitted informally to Mr. Dominic DiIanni of your staff on February 20, 1979, in response to your informal transmittal of February 9, 1979. In that your letter of February 16, 1979 was not received by Met-Ed until February 26, 1979, and consistent with discussions with your Mr. DiIanni, Met-Ed's responses are being submitted on this date.

Also enclosed, please find a revised Page 2-2 of the TMI-1 Technical Specifications. Page 2-2, which was inadvertently omitted from the original submittal of December 28, 1978 (GQL 2068), has been revised to include the new values for FΔH and Fq.

Sincerely,

J. G. Herbein
Vice President - Generation

JGH:RJS:clb

Enclosures

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TMI-1 CYCLE 5 RELOAD

Additional Information

1. NRC CONCERN

The control rod group withdrawal anticipated operational occurrence analysis shown in the FSAR is based (nominal conditions) on assumed values of the moderator and doppler temperature coefficients less adverse than values of these coefficients predicted to occur during the forthcoming cycle. Sensitivity studies shown in the FSAR show the effect of the more negative doppler coefficient but do not span the range of the anticipated moderator temperature coefficient. Please explain how the system trip setpoints afford plant protection (DNBR and RCS pressure) at these more adverse conditions. Please consider the full range of bank worths in your analysis.

MET-ED RESPONSE

Table 7-1 of the TMI-1 Cycle 5 Reload Report indicates that the BOC values (worst conditions for the rod withdrawal accident) for the FSAR Moderator and Doppler reactivity coefficients are less negative than the Cycle 5 values. Although Figures 14-12 and 14-13 of the FSAR indicate that the reactivity coefficients create higher pressures, it must be recognized that these figures are based on a sensitivity study at one given rod worth and may not be indicative of a pressure versus reactivity relationship at another rod worth. The important relation is the peak pressure versus rod withdrawal rate, shown in Figure 14-10, which spans the range of rod worth and reactivity feedback combinations, encompassing the reactivity versus power and pressure relations of the Cycle 5 Reload. Changes in the reactivity coefficients shift the point where the peak pressure occurs with respect to rod withdrawal rate and/or move the range of single to all control rod groups (based on the nominal values listed

in Tables 14-3, 14-5, and 14-6) along the curve. The magnitude of the peak pressure will not change, and therefore, the design over power and/or RCS pressure limits will not be exceeded. Since the Cycle 5 core parameters do not result in more adverse conditions than the worst cases studies in the FSAR, the system trip setpoints do provide plant protection for DNB and RCS pressure concerns.

2. NRC CONCERN

The dropped rod anticipated operational occurrence analysis shown in the FSAR is based on a doppler coefficient less adverse than the value predicted for the forthcoming Cycle 5. Explain why you consider the FSAR analysis bounding for Cycle 5. Provide the post rod drop peak enthalpy rise assumed in the FSAR analysis and predicted for Cycle 5.

MET-ED RESPONSE

As noted in the response to NRC CONCERN 1, the severity of the transient is less dependent of doppler and moderator (moderator even less so, due to the rapid insertion of reactivity) reactivity coefficients than on the maximum worth of the dropped rod. From Table 7-1 of the Cycle 5 Reload Report comparing the dropped rod worth for the FSAR analysis of 0.46% $\Delta k/k$ with the Cycle 5 value of 0.2% $\Delta k/k$, the factor of more than 2 times in difference between the two values indicates by inspection that the FSAR case will bound Cycle 5 for small variations in the reactivity coefficients. The post rod drop peak enthalpy rise is not an FSAR requirement. Babcock and Wilcox has not performed this analysis for any of its plants.

3. NRC CONCERN

The ejected rod accident analysis presented in the FSAR is based on values of B eff less adverse than predicted for Cycle 5. Explain why you consider the FSAR analysis bounding for Cycle 5. Furthermore, confirm that the post ejected value of the peak linear heat rate assumed in the FSAR analysis bounds the Cycle 5 predicted values.

MET-ED RESPONSE

Although the ejected rod accident analysis presented in the FSAR is based on larger β eff values (0.0071 vs. 0.0058), the FSAR analysis is still bounding for Cycle 5. This is a result of the larger worth (0.65% $\Delta k/k$) assumed for the ejected rod in the FSAR analysis, as compared to 0.25% $\Delta k/k$ for Cycle 5. This is best illustrated if the reactivity is expressed in terms of dollars = ρ/β. For the FSAR case, this is \$ = .0065/.0071 = ρ/β 0.92. For Cycle 5, reactivity added is \$ = .0025/.0058 = 0.43, more than a factor of 2 less than the FSAR. The post ejected value of the peak linear heat rate assumed in the FASR analysis bounds the Cycle 5 predicted values, since the design peak is used for the FSAR analysis and the Cycle 5 peaks have been shown to be less.

4. NRC CONCERN

Please confirm the applicability of the BAW 1461, "Reactivity Insertion Assumptions used In Safety Analysis Calculations", to the analysis of TMI-1, Cycle 5. If applicable, show that sufficient margin will exist during Cycle 5 to accommodate the 0.09 DNBR reduction during the hypothetical four-pump coastdown sited in BAW 1461.

MET-ED RESPONSE

The topical report, BAW-1461, "Reactivity Insertion Assumptions used In Safety Analysis Calculations", is applicable to TMI-1, as noted in the introduction of the report. Since the 4-pump coastdown assumptions are based on the worst case BOC conditions, Cycle 5 and all subsequent cycles are bounded by the information contained in the report. Sufficient margin will exist during Cycle 5 to accommodate the 0.09 DNBR reduction during the hypothetical 4-pump coastdown, because the minimum DNBR from Table 6-1 of the Cycle 5 reload report is given as 1.98. This would provide Cycle 5 with sufficient margin to accommodate the 0.09 (9 point) DNBR reduction, as did the bounding case in the topical.

5. NRC CONCERN

Please confirm that the clad collapse calculation for Cycle 5 were performed using the CROV computer code and associated standard modeling techniques.

MET-ED RESPONSE

The creep collapse analysis was performed based on the CROV code topical BAW-10084, Rev. 1. (See TMI-1 Cycle 5 Reload Report, Reference 3).

6. NRC CONCERN

Provide the analytic bases for the revision of Technical Specification 3.2.2 which would increase the minimum boric acid mix tank level from 800 $ft.^3$ to 906 $ft.^3$.

MET-ED RESPONSE

Boric acid storage volumes required for RCS boration to cold shutdown are sensitive to fuel cycle physics parameters such as doppler deficit, moderator deficit, total and stuck rod worths, fuel enrichment, and batch size. These parameters change on a cycle-to-cycle basis, thereby affecting the boron concentration requirements for shutdown. The shutdown calculations require a 1% $\Delta k/k$ shutdown margin with no xenon and highest worth stuck rod. The increased boric acid volume for TMI-1 Cycle 5 relative to Cycle 4 is a direct result of increased boron concentration requirements imposed by fuel cycle differences between the two cycles.

7. NRC CONCERN

Cycle 5 values of permitted axial power shape rod (APSR) position vs. core power, shown as proposed Figure 3.5-2H of the plant Technical Specifications, will require long term insertion of the APSR during rated power production. The APSR is to be withdrawn no less than 6.1%, nor no more than 45%, during operation of greater than 92% of rated power. Please provide predicted values of $F\Delta H$ and Fq following long term operation with the APSR's 6.1% withdrawn and subsequent withdrawal of the APSR's to 45% withdrawn.

MET-ED RESPONSE

The APSR position at full power at which the core offset will be minimized is approximately 30% withdrawn. This configuration will be maintained for long term steady state operation. The limiting APSR positions would only be approached for the control of short-term, transient axial effects. The following table gives the values of Fq (peak pellet) and $F\Delta H$ (peak

pin) predicted for the nominal and limiting APSR positions for the end of Cycle 5 after long term operation with the APSR's at 32% withdrawn. The values below are nominal, no uncertainties have been added.

APSR Position %WD	$F_{\Delta H}$ (Location*)	F_q (Location*)
6.1	1.27 (K-11)	2.17 (L-12)
32	1.28 (K-11)	1.54 (L-14)
45	1.28 (K-11)	1.93 (L-12)

*Locations are 1/8 core symmetric

It should be noted that withdrawl of the APSR's to 45% WD without movement of Bank 7 from its nominal position (287% WD Rod Index) will produce an imbalance of -18.8%, which is outside of and would be precluded by the imbalance limits of proposed Technical Specification Figure.

8. NRC CONCERN

Figure 5-1 of your Cycle 5 Reload Report shows the beginning of cycle predicted planar power distribution with the APSR's inserted. Does this calculation (a two dimensional PDQ07 calculation) represent the APSR's as if they were full length, full strength rods, or have cross sections been adjusted to represent the reduced length of the APSR's?

MET-ED RESPONSE

The two dimensional PDQ07 calculations represent the reduced length of the APSR's by having flux and volume weighted cross sections for the APSR's pins from a three dimensional PDQ07 calculations which had the

APSR's shown explicitly. Further normalization of the two dimensional model to the three dimensional analy s is done by applying an increased axial leakage (buckling) to the fuel assemblies containing the APSR's. This model correctly accounts for the radial peaking and assembly average burn up for the assemblies containing APSR's in the two dimensional calculations.

9. NRC CONCERN

Please confirm the applicability of BAW-10121P, "RPS Limits and Setpoints", to TMI-1, Cycle 5.

MET-ED RESPONSE

Topical Report BAW-10121P, "RPS Limits and Setpoints", was written specifically to address RPS - II type plants. TMI-1 is an RPS-1 plant, and therefore, the information contained in the report does not apply to TMI-1. The techniques used to determine the RPS setpoints for TMI-1 are outlined in Section 2.3, Bases, of the TMI-1 Technical Specifications.

10. NRC CONCERN

Please provide the quantitative, rather than qualitative, bases for your revision of the bypass flow to 10.4% of total flow to accommodate the effect of orifice rod assembly removal.

MET-ED RESPONSE

A detailed review of the methods used to calculate guide tube leakage was conducted by the NRC staff in conjunction with the review of Davis - Besse Cycle 1 operation with BPRA's and ORA's removed[1]. This review

resulted in a may 1978 meeting in Bethesda between NRC's Mr. M. W. Hodges and B&W's Mr. G. A. Moyer. The NRC approved the guide tube leakage calculation at that time. Since then, B&W has used the same method to license Oconee I[2], Oconee II[3], Oconee III[4], Crystal River 3[5], and Rancho Seco[6].

11. NRC CONCERN

Please provide the quantitative, rather than qualitative, bases of your review of the peak enthalpy rise, FΔH, from 1.78 to 1.71 to accommodate the revised bypass flow.

MET-ED RESPONSE

The removal of orifice rods in Cycle 5 will increase core bypass flow by 2%. To offset the reduction in core flow, credit was taken for some of the large margin between the calculated Cycle 5 radial x local peak of 1.403 and the previously used reference design peak of 1.78 (FΔH).
A value of 1.71 was chosen for FΔH. The primary impetus for the use of this value was that it had previously been reviewed and approved by NRC for use in the thermal hydraulic design of Davis-Besse, Cycle 1[7], before the concern with BPRA and ORA latching mechanisms. Therefore, when it became desirable to remove ORA's and/or BPRA's, precedent had already been set for the 1.71 value. Other B&W plants which have had FΔH reduced from 1.78 to 1.71 are Oconee I Cycle 5[2], Oconee II Cycle 4[3], Oconee III Cycle 4[4], Rancho Seco 3[6], and Crystal River 3 Cycle 1[5]. Experience has shown that 1.71 is more realistic than 1.78 but still provides conservative margins to steady state and maneuvering peaking limits.
Transient analysis is begun from initial conditions at 102% power. The minimum DNBR at 102% power has increased from 2.24 to 2.33 in going from

the 1.78 to 1.71 radial x local peak. The transient analysis applicable to cycle 4 is conservative for cycle 5 because of the increase in initial minimum DNBR.

The limiting flow transient for TMI-1 is the one pump coastdown which determines the flux/flow trip setpoint. The cycle 5 flux/flow setpoint is 1.08 for TMI-1. The minimum DNBR during the one pump coastdown with this setpoint is 1.74 based on the 1.71 value of FΔH. This leaves 20% margin to the minimum DNBR criteria for cycle 5 which is 1.43 with 11.2% rod bow penalty.

12. NRC CONCERN

Are Figues 8-1 and 8-2, Core Protection Safety Limits, Trip Setpoint for Nuclear Overpower Based on RCS Flow and Axial Power Imbalance, respectively, based on an assumed FΔH of 1.71 or 1.78?

MET-ED RESPONSE

Figues 8-1 and 8-2 are based on an assumed FΔH of 1.78.

13. NRC CONCERN

Table 1 of your submittal shows that safety limits calculated for Cycle 5 are less restrictive than the proposed Technical Specifications Safety Limits (SL). By inference you assert that the Limiting Safety System Setpoints (LSSS) corresponding to Technical Specification SL are more restrictive than the LSSS that would correspond to the Cycle 5 SL. Please confirm this assertion. Consider transient DNBR degradation during the course of postulated transients for which the LSSS are to provide protection, as well as steady state conditions used to determine the SL.

MET-ED RESPONSE

The proposed Technical Specifications Safety Limits (Table 1 of the submittal) are based on the Limiting Safety System Setpoints from the latter part of Cycle 4 (Figure 8-2 of the Cycle 5 Reload Report). These limits were determined for the Standard Tech. Specs. by providing the most restrictive envelope, such that all future cycles would be bounded. Unlike previous cycles, the Standard Tech. Spec. Safety Limit envelope was directly calculated from the limits of the Cycle 4 trip setpoint envelope, thereby providing a more restrictive Safety Limit envelope but allowing greater variations in the offset Limits for subsequent cycles. Therefore, as noted in the question, the LSSS corresponding to the Tech. Spec. SL are more restrictive than the LSSS corresponding to the Cycle 5 SL noted in Table 1 of the referenced submittal.

With respect to DNBR degradation during the course of postulated transients, the flux/flow trip setpoints provide protection to maintain adequate margin for DNB. The "winged" portions of each pump operation envelope provide adequate margin to MDNBR for steady state conditions.

14. NRC CONCERN

Please commit to provide a startup test report.

MET-ED RESPONSE

Within 90 days following completion of TMI-1's Cycle 5 startup and physics testing, a startup and test report will be submitted to NRC.

REFERENCES

1. Attachment 1 to Application to Amend Operating License for Removal of Burnable Poison Rod and Orifice Rod Assemblies, BAW-1489, Rev. 1, May 1978.

2. Oconee I Cycle 5 - Reload Report - BAW-1493, Rev. 2, September 1978.

3. Oconee II Cycle 4 - Reload Report - BAW-1491, August 1978.

4. Oconee III Cycle 4 - Reload Report - BAW-1486, Rev. 1, June 1978.

5. Crystal River Unit 3 - Licensing Considerations for Continued Cycle 1 Operation Without Burnable Poison Rod Assemblies, BAW-1490, Rev. 1, July 1978.

6. Rancho Seco Nuclear Generating Station, Unit I - Cycle 3 Reload Report - BAW-1499, September 1978.

7. Davis - Besse Nuclear Power Station, FSAR.

The elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112 percent) when the reactor coolant flow is $139.8 \times 10^{+6}$ lbs/h, which is less than the actual flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (2) with potential fuel densification and fuel rod bowing effects;

$$F_q^N = 2.57, \quad F_{\Delta H}^N = 1.71; \quad F_z^N = 1.50$$

The 1.5 axial peaking factor associated with the cosine flux shape provides a lesser margin to a DNBR of 1.3 than the 1.7 axial peaking factor associated with a lower core flux distribution. For this reason the cosine flux shape and the associated $F_z^N = 1.50$ is more limiting and thus the more conservative assumption.

The 1.50 cosine axial flux shape in conjunction with $F\Delta H = 1.71$ define the reference design peaking condition in the core for operation at the maximum overpower. Once the reference peaking condition and the associated thermal-hydraulic situation has been established for the hot channel, then all other combinations of axial flux shapes and their accompanying radials must result in a condition which will not violate the previously established design criteria on DNBR. The flux shapes examined include a wide range of positive and negative offset for steady state and transient conditions.

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing;

a. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.57$ of the combination of the radial peak, axial peak, and position of the axial peak that yields no less than 1.3 DNBR.

b. The combination of radial and axial peak that prevents central fuel melting at the hot spot. The limit is 19.6 kW/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.