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March 7, 1983
5211-83-063

Office of Nuclear Reactor Regulation
Attn: John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing
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
Pressurized Thermal Shock (PTS)
Flux Reduction

As a result of our meeting with your staff on January 14, 1983, we committed to supply you with information available concerning questions raised by the Staff on flux reduction. The enclosed information addresses those specific questions.

As a step in the direction of resolving PTS, we have joined with other B&W Owners to form a PTS Task Force within the B&W Owners Group, to continue the dialogue begun during the week of January 10. During those meetings, we presented the status of several key programs that deal with confirming the continued integrity of B&W vessels not only for PTS events but in accordance with requirements of 10CFR50 Appendices G&H. We believe that once the NRC Staff has a chance to review our programs in depth, they will concur that the risk of PTS to B&W vessels is acceptably low and within the limits applied to the rest of the industry.

Furthermore, we believe that a number of alternatives to reaching the screening criterion should be allowed in considering a proposed rule and we are prepared to technically substantiate these alternatives within the next several months. Among the alternatives are:

- Improvement in chemistry characterization for critical materials.
- Refinement of uncertainties in initial RTndt values.

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- Development of RTndt shift correlation for B&W weld metal.

- Refinement of vessel fluence calculations to further reduce uncertainty.

At TMI, we are pursuing flux reduction schemes and are committed to resolving the PTS issue. We (as a member of the B&W PTS Task Force) will be meeting with the NRC Staff regularly over the next several months. We expect to review a number of other PTS related actions to clarify or mitigate our mutual concerns in addition to flux reduction.

Sincerely,

H. D. Hukill
Director - TMI-1

HDH/klk
Enclosure

cc: J. Van Vliet
R. Woods
R. Conte

ENCLOSURE

NRC Question 1:

Provide your assessment of the fluence experienced to date by the welds and plates in your pressure vessel, the rate of increase expected assuming future fuel cycles to which you are already committed, and a detailed description of the bases for the above (including surveillance capsule date and analysis methods, and generic methods or correlations used).

Response 1:

An assessment of fluence experienced to date by the TMI-1 pressure vessel is included in Reference 1 (TMI-1 PTS Report) which was submitted in June 1982. Since then the plant has remained shut down per NRC order and no additional fluence has accrued. As stated in the report (Section 7.3), TMI-1 has completed four cycles of operation corresponding to 3.52 EFPY with a total neutron fluence of 0.18×10^{19} n/cm^2 at the peak location. The report also includes a description of the bases for this fluence value, including surveillance capsule data, analytical methods, generic evaluations, and correlations used.

Cycles 1 to 4 utilized an annual out-in-in fuel management strategy. The calculated maximum fluence accumulation rate using this scheme for future cycles would be 0.055×10^{19} n/cm^2 per EFPY. Cycle 5 is also designed as an out-in-in cycle. The projected cycle capacity factor for Cycle 5 is about 73% and the expected total fluence accumulation at the end of cycle is about 0.22×10^{19} n/cm^2.

Current fuel management planning is for conversion to an in-out-in fuel loading strategy beginning with Cycle 6. Vendor (B&W) estimates to date, which are based on similar extended-cycle designs for other 177-fuel assembly plants, show that the in-out-in scheme will provide a reduction in peak fluence of approximately 30%. This will result in an incremental fluence accumulation rate of about 0.039×10^{19} n/cm^2 per EFPY for TMI-1. Based on current projected generation schedules and using a cumulative capacity factor of 73% for Cycles 6 and beyond, the total peak fluence at end-of-license (EOL), which occurs in 2008, is estimated as 0.89×10^{19} n/cm^2.

NRC Question 2:

Using the above fluence information, provide your assessment of the RT_{NDT} presently existing in your pressure vessel welds and plates utilizing the methodology outlined in Appendix E to Enclosure A of SECY-82-465, and the expected future rates of increase, and the expected dates when the applicable proposed screening criterion will be exceeded.

Response 2:

The table below summarizes the requested information. A cumulative capacity factor of .73 for Cycle 5 and beyond was used.

Date	Δ Fluence 10^{19} n/cm² per EFPY	Peak Fluence 10^{19} n/cm²	RT$_{NDT}$ (°F) R. G. 1.99 circ WF 70	R. G. 1.99 circ WF 25	R. G. 1.99 axial WF 8	R. G. 1.99 axial SA 1526	Guthrie +2σ circ WF 70	Guthrie +2σ circ WG 25	Guthrie +2σ axial WG 8	Guthrie +2σ axial SA 1526
07/15/83 (Current)	0.051	0.18	122	140	119	131	192	209	182	202
07/15/84 (EOC-5)	0.055	0.22	134	155	131	142	199	217	189	210
October/ 2003 (SA-1526 reaches 270°F)	0.039	0.76	254	268	244	259	255	280	240	270
05/18/08 (End of License)	0.039	0.89	262	277	264	267	263	290	248	279
Screening Criteria			300°F		270°F		300°F		270°F	

Allowable fluence at the controlling weld (SA-1526) using the Guthrie correlation is 0.64×10^{19} n/cm² to meet the 270°F RT$_{NDT}$ axial weld screening criteria.

NRC Question 3:

Provide a description of the flux reduction options that you have considered for your plant. Include for each option:

a. Description of fuel management and/or fuel removal and/or fuel replacement with dummy elements, showing core maps for future cycles;

b. Quantitative assessment of resulting flux reduction to critical welds and plates;

c. Parametric study showing future RT$_{NDT}$ values resulting from both the earliest practicable implementation of the option, and from the latest possible implementation of the plan that will still avoid exceeding the RT$_{NDT}$ screening criterion at end-of-life;

d. Discussion of advantages and disadvantages of the option, particularly emphasizing power reductions caused by the option. With respect to power reduction, discuss the magnitude of the reduction and the particular limit (e.g., hot channel factor, DNBR, etc.) causing the power reduction. Also analyze how much relief would be necessary (with respect to the particular limit) to allow full power operation, and assess whether such relief would be an improvement to overall plant safety (considering LOCA, PTS, transients, etc.).

Response 3:

3.1 In-Out-In Strategy

a. Present planning includes conversion from the current annual out-in-in fuel management strategy to an extended-cycle in-out-in loading scheme which will reduce the neutron fluence to the reactor vessel. This change will be implemented with Cycle 6. Figure 1 (attached) shows a 1/8-symmetric core map for Cycle 5, an annual cycle, indicating the locations of fresh, once, twice, and thrice-burned fuel assemblies. Figure 2 (attached) shows similar preliminary core maps for transition Cycles 6 and 7 using a 76 fresh fuel feed batch and once and twice-burned assemblies; burnable poison rods will be used. Figure 3 (attached) shows preliminary core maps for a 72 feed equilibrium 18 month cycle which will also utilize burnable poison rods.

b. Vendor (B&W) estimates based on similar extended-cycle designs for other 177-FA plants indicate that the in-out-in fuel management scheme will provide a reduction in peak fluence of approximately 30% below that for the current out-in-in scheme.

Fluence reduction factors will be known more accurately when analysis of the final design of Cycle 6 is completed later in 1983 and specific power and flux distributions are available, however, estimated end-of-license fluence values are shown in the following table:

		- - - - End-of-License Fluence -- $10^{19} n/cm^2$ - - - -		
Weld Designation	Weld Factor	With Unmodified Fuel Management Plan	With Modified Fuel Management Plan	To meet Screening Criteria
WF 70	0.76	.90	0.68	1.25
WF 25	1.00	1.18	0.89	1.05
WF 8	1.00	1.18	0.89	1.34
SA 1526	0.84	0.99	0.75	0.64

c. The earliest practicable implementation of the in-out-in fuel management scheme is Cycle 6, presently scheduled to start in October 1984.

As indicated in the chart of Response #2, this plan does not result in all welds meeting the RT_{NDT} screening criteria by the end-of-license using the Gurhrie equation. The SA-1526 axial weld RT_{NDT} is calculated to exceed the 270°F screening criterion in 2003 after an additional 14.6 EFPY, based upon current TMI-1 generation forecasts.

At end-of-license, in 2008, the Guthrie correlation predicts an RT_{NDT} of less than 10°F above the 270°F criterion. All other welds of concern (WF 70, WF 25, WF 8) remain below the screening criteria at EOL using the Guthrie correlation.

It may be noted that all welds, including SA 1526, remain below the screening criteria when the correlations of Regulatory Guide 1.99 are used.

d. Based on B&W extended-cycle designs for other 177-FA plants the in-out-in strategy for TMI-1 is not now expected to result in any core power reductions or decreases in operating and maneuvering limits.

3.2 In-In-Out Strategy

a. GPUN is investigating other fuel cycle design options that could be used to decrease reactor vessel accumulated fluence. These options are based on an in-in-out, or very low leakage (VLL), extended cycle fuel management scheme.

The VLL scheme is an extension of the in-out-in (LBP) design in that it further reduces RV fluence by reducing the relative power of the peripheral fuel assemblies. Most of the fast fluence seen by the RV is due to the peripheral fuel assemblies. Fluence reduction is accomplished with the LBP design by placing once-burned assemblies on the core periphery with fresh fuel loaded in the interior core regions. The VLL scheme places twice-burned assemblies on the periphery thus further lowering the power of the peripheral assemblies. Burnable poison rods are also used in the VLL design. A schematic of the various fuel management schemes is shown in Figure 4.

b. A moderate estimate of fluence reduction for the VLL scheme is 20% over that of the in-out-in scheme. This would give an incremental peak fluence value of $0.031 \times 10^{19} n/cm^2$ per EFPY for TMI-1. Using a 73% cumulative capacity factor for Cycle 6 and beyond, and assuming the VLL scheme is implemented for Cycle 6, an EOL peak fluence of $0.75 \times 10^{19} n/cm^2$ results.

c. This fluence results in an RT_{NDT} at the controlling weld (SA-1526) of 269°F which meets the 270°F axial weld screening criterion.

A scoping study to evaluate the VLL scheme was recently completed by B&W. These results are not yet available for use by GPUN. However, preliminary information indicates that reduction in the incremental fluence factor may be greater than the 20% estimate.

d. Based on the B&W scoping study, use of the VLL scheme is expected to increase core power peaking by several percent. Plant-specific analyses would be required to establish new operating limits and to verify the feasibility of the VLL scheme for TMI-1.

3.3 Further Studies

The estimates discussed above indicate that TMI-1 would meet the present screening criteria with the in-in-out (VLL) scheme. The criteria may also be met with the currently planned in-out-in fuel loading scheme, when detailed core design analyses are performed for TMI-1.

Nevertheless, GPUN is considering funding additional analyses to more precisely determine the neutron flux contribution from each peripheral fuel assembly location which could be used to refine fuel loading patterns.

GPUN is also participating in the "Fluence Analysis and Dosimetry" tasks of the RV Materials Evaluation Program. These tasks are expected to provide more accurate RV fluence determinations with minimal uncertainties and are discussed further in Response 4 below.

NRC Question 4:

Discuss any alternatives you may be considering to flux reduction that will result in delaying or avoiding exceeding the RT_{NDT} screening criterion. These would include topics such as archival materials research, plans to sample and analyze as-built materials, etc.

Response 4:

GPU Nuclear, as a member of the B&W Owners Group Materials Subcommittee, is involved in several programs that may provide acceptable alternatives to flux reduction. They include:

4.1 The B&W Owners Group has previously funded a program to more fully characterize the chemistry of reactor vessel weld metal. This effort consisted of chemical analysis of both weld metal surveillance program samples and actual reactor vessel archive weld metal. The results of this study, documented in Babcock & Wilcox report BAW-1500, September 1978, will aid in reducing uncertainties in RT_{NDT} calculation due to variations in weld metal chemistry.

4.2 Current studies by B&W are aimed at characterizing the initial RT_{NDT} of B&W weld metals. A statistically significant data base for these materials exists. The results of this analysis will serve to better define both the initial RT_{NDT} value and the variation associated with it.

4.3 The B&W Owners Group, under the RV Integrity Program, has obtained a substantial body of data on the RT_{NDT} of irradiated B&W beltline weld metals. A B&W correlation has been developed that predicts significantly lower shifts of RT_{NDT} as well as a significantly reduced band of uncertainty.

Current plans of the Owners Group include the submittal of a topical report during April of 1983. We believe that the use of this refined correlation as an alternative to the methodology specified in SECY 82-465, will demonstrate that the TMI-1 RV welds will not exceed the proposed screening criterion, using planned fuel management strategies, during the life of the plant. The use of this methodology is consistent with 10CFR Appendices G&H as well as the Commission's position of allowing alternate means of not exceeding the screening criteria.

4.4 The Owners Group is currently working with B&W on the feasibility of benchmarking calculated vs. actual pressure vessel fluence. Conceptually, this effort would consist of installing dosimeters both inside and outside the vessel at the same location, and additional dosimeters at the same elevation over part of the vessel outside circumference.

The results of this effort would enable B&W Owners to more precisely define the inner vessel wall fluence and its variation with circumferential location. Since the critical TMI-1 weld is not at the calculated location of highest fluence, a more precise definition of the fluence at this location would reduce the uncertainty associated with the current RT_{NDT} calculations.

4.5 To further improve the accuracy of RV fluence determinations GPUN is participating in Phase XA (Fluence Calculation Benchmarking) of the Reactor Vessel Materials Evaluation Program. This task will benchmark analytical techniques by comparing calculated fluence values to a known flux source in a simulated 177-FA configuration with dummy capsules. Expected benefits are a reduction of fluence error bands due to uncertainty and a corresponding reduction in predicted material degradation.