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September 20, 1982

5211-82-221

Office of Nuclear Reactor Regulations
Attn: Mr. John F. Stolz, 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
Schedule for Completion of NUREG 0737
Item II.B.2 (Plant Shielding)

This letter is in further response to your letter dated August 20, 1982 which requested revised commitments and/or justification of completing NUREG 0737 Items beyond restart.

In our response to Generic letter 82-05 (5211-82-109, May 21, 1982 and 5211-82-144, June 15, 1982) we indicated that the plant shielding modifications involving remote valve operators for certain Decay Heat Removal (DHR) system valves would not be completed until startup from Cycle 6 refueling. Justification for this schedule is discussed below.

Positioning of valves DH-V12A/B, DH-V38A/B, and/or DH-V19A/B provides a positive means (i.e. forced flow) of preventing post accident boron precipitation. The flow paths provided by operation of these valves are: (1) flow through the core and out the DHR drop line (flow through the core would occur without operator action for hot leg breaks); or (2) forced flow into the hot leg via the pressurizer spray line or DHR drop line causing reverse flow through the core for cold leg and core flood line breaks. Failure of DH-V64, RC-V4, or DH-P1A would prevent implementing method (2) above via the pressurizer auxiliary spray line. Furthermore, considering core damage to the extent required by NUREG 0737 item II.B.2, neither hot leg injection via DHR drop line nor draining via the DHR drop line could be implemented to overcome these single failures due to high radiation. Such an event with fuel damage beyond the design basis is considered to be of very low probability.

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B&W has studied internal reactor vessel circulation in B&W 10091 supplement 1 and has concluded in BAW 10103 Rev. 3 that "post-LOCA natural circulation through the leakage gaps between the outlet nozzles and core support shield will preclude the possibility of unacceptably high boric acid concentration in the core region during the long-term cooling phase following the postulated LOCA". On this basis, it is questionable if any manual valve positioning is needed. Nevertheless, we believe that on a realistic basis, the abovementioned flow paths could be implemented. The post accident radiation levels realistically projected would permit operator access to perform the manual valve alignments.

For a core flood line break core uncovery is not predicted, therefore, no core damage above a hypothetical pre-existing 1% failed fuel assumption would exist. This level is much less than would be expected for a large break LOCA. For a worst case large break LOCA it is expected that the fuel gap activity would be released but that no other major release due to clad melting or significant cladding oxidation would occur. As discussed in the attached (Attachment 1) evaluation, the projected exposure to an operator during positioning of the valve in the highest radiation area using realistic methods is about 1 Rem. On this basis we believe that the specified manual valve alignments could realistically be accomplished and that completion of the remaining modifications to address NUREG 0737 II.B.2 can be postponed until Cycle 6 refueling without undue risk.

Finally, a chronology of our efforts to date and a projection of engineering/procurement to complete these modifications is also attached (Attachment 2).

Sincerely,

H. D. Hukill
Director, TMI-1

HDH:CWS:vjf
Attachments
cc: R. Jacobs
R. C. Haynes

EVALUATION OF POST LOCA RADIATION LEVELS ASSOCIATED WITH MANUAL OPERATION OF DECAY HEAT REMOVAL SYSTEM VALVES TO PREVENT BORON PRECIPITATION

I. Basic Assumptions

A. 100% of the gap activity is released to the Reactor Coolant (TMI-2 FSAR Chapter 15A Table 2).

B. 1% failed fuel exists before the LOCA with resulting reactor coolant activity (TMI-1 FSAR Update Chapter 14).

C. No clad melting or signicant cladding oxidation occurs for the LOCA (TMI-1 FSAR Update Chapt. 14).

D. The Decay Heat Removal System is recirculating depressurized Reactor Coolant, therefore, no noble gas inventory was assumed to be recirculated.

E. Access to the DHR valve area is not attempted until 24 hours after the LOCA. (Operating Procedure 1104-4)

F. The time for the operator to proceed through the area and realign the valves is 5 minutes. (Restart Report Section 2.1.2.3)

G. Other assumptions are the same as those in TMI-1 FSAR Update Chapter 11 Appendix A.

II. Analysis

A. From the TMI-2 FSAR Chapter 15 Appendix A Table 2 the fuel gap fission product inventory at the end of 930 days of power operation was obtained. (Major contributor is iodine).

B. The gap activity for TMI-1 was then derived by multiplying the ratios of the thermal power levels $\frac{(2535)}{(2772)}$

C. The recirculation concentration (Table 1) was calculated using a primary recirculation volume of 1.85×10^9cc. (The same volume as used for the NUREG 0737 II.B.2 analysis.)

The "activity factor" was then derived by comparing recirculation concentration of the realistic analysis to those performed in accordance with NUREG 0737.

$$\lambda_A = \frac{\sum \left[{}^{Ci}/_{cc}\right]_{\text{Real}}}{\sum \left[{}^{Ci}/_{cc}\right]_{\text{NUREG}}} = 6.48 \times 10^{-3}$$

Additionally an energy weighted "activity factor" was also derived by comparing the recirculation concentrations of the two analyses.

$$\lambda_E = \frac{\sum \left[\left({}^{Ci}/_{cc}\right) * E_y\right]_{\text{Real}}}{\sum \left[\left({}^{Ci}/_{cc}\right) * E_y\right]_{\text{NUREG}}} = 2.83 \times 10^{-3}$$

Worst case activity factor was λA.

The NUREG 0737 analysis calcuated a dose rate of 20,000 rem/hr for t=0.

This dose rate was reduced by a decay factor (F) to adjust T=24 hrs.

$$DR_{24} = DR_0 \times F = 20{,}000 \text{ Rem/hr.} \times 0.09 = 1800 \text{ Rem/Hr.}$$

Given that the evolution will take 5 min. for an operator to perform, the evolutions following dose is obtained.

$$D_{\text{NUREG}} = 1800\ {}^{r}/_{hr} \times 5 \text{ min.} \times \frac{1 \text{ hr}}{60 \text{ min}}$$

$$D_{\text{NUREG}} = 150 \text{ Rem}$$

Applying the above, worst case activity factors (λA) to the NUREG 0737 calculation, a more realistic dose is determined.

$$D_{\text{real}} = D_{\text{NUREG}} \times \lambda A$$

$$= 150 \text{ Rem} \times 6.5 \times 10^{-3}$$

$$D_{\text{real}} = .975 \text{ Rem}$$

T A B L E   I

ISOTOPE	RECIRCULATION CONCENTRATION µCi/cc TMI-1*	RECIRCULATION CONCENTRATION FROM NUREG 0737 ANALYSIS µCi/cc
I-131	6.98×10^2	1.99×10^4
-132	100.9	2.33×10^4
-133	151.7	3.45×10^4
-134	9.5	4.32×10^4
-135	48.1	3.43×10^4
Cs-134	6.27	6.85
-136	0.28	4.33
-137	22.9	2.69×10^1
-138	0.144	6.64×10^2
Ba-137m	13.5	2.52×10^1
-140	0.229	6.75×10^2
La-140	0.225	6.85×10^2
Ce-144	0.0135	4.05×10^2

*Includes 1% failed fuel

Plant Shielding Review - Decay Heat Removal Valve Operators

Chronology

July 1979

NUREG 0578 (Item 2.1.6.b) issued to "assure that access and performance will not be unduly impaired due to radiation from these systems".

September 13, 1979

NRC letter on follow up actions resulting from NRC Staff reviews regarding TMI-1 accident.

October 30, 1979

NRC letter on lessons learned short term requirements invoking GDC 19 dose rate criteria.

November 13, 1979

Met Ed contracts with Gilbert Associates to perform a plant shielding study for TMI-1.

May 1980

NUREG 0660 NRC Action Plan Item II.B.2 issued.

June 1980

SER (NUREG 0680) item 2.1.6.b indicated NRC awaiting TMI-1 plant shielding study.

September 5, 1980

NRC letter clarifying TMI Action Plan requirements.

October 2, 1980

Met Ed letter indicates that the plant shielding study will be provided by October 15, 1980 and modifications to correct any problems will be described by June 30, 1981.

October 17, 1980

Restart Report Amendment 22 issued containing TMI-1 Plant Shielding study.

October 31, 1980

NRC Letter endorsing the requirements of NUREG 0737 which provides additional clarification for II.B.2. Because of industry procurement problems grants an extension for modifications to January 1, 1982 or the first outage of sufficient duration after but no later than July 1, 1982.

January 23, 1981

Met Ed letter in response to NUREG 0737 indicating the 1st outage through July 1982 for completion of plant shielding mods.

February 13, 1981

Met Ed letter to the Commissioners indicating a projected criticality date of October 1981.

February 1981

Engineering effort begins on DHR valve remote operators.

April 1981

SER Supplement 3 indicates reasonable progress on II.B.2.

May 13, 1981

Restart Report Amendment 25 provides revised shielding study and describes proposed modifications for the DHR system including remote operators for DHV 12 A/B, 19 A/B and 38 A/B. (DH-V64 protected by shield wall to be erected before Restart.)

August 1981

Bid specifications for the 6 DHR valves issued.

November 1981

Met Ed identifies OTSG tube leak problems.

December 16, 1981

Proposal made by Walworth on DHV 12 A/B, and 38 A/B.

January 1982

For resource conservation to focus on OTSG work, II.B.2, among other things, was placed on engineering hold.

January 25, 1982

Met Ed announced a minimum of six month delay in Restart due to OTSG problems.

March 16, 1982

Comments sent to Walworth concerning problems with their proposal.

May 15, 1982

Plant shielding taken off of engineering hold.

May 21, 1982

GPUN response to Generic Letter 82-05 identified procurement problems with vendors and requested an extension to Cycle 6 refueling.

June 15, 1982

GPUN letter further amplifies problems associated with the engineering/bid evaluation process concerning DHR remote operators.

June 30, 1982

Pre-award meeting with Walworth.

PROPOSED SCHEDULE FOR DHR VALVE OPERATOR MODIFICATION

September 20, 1982

Specification to be con. ɔrmed for valve operators from Walworth.

October 1, 1982

Place purchase order for valve operators.

October 15, 1982

Pre-award meeting with CCI (DHV 19 A/B).

December 1982

Complete purchase order on safety grade control panel.

April 1982

Delivery of Remote Valve Operators (Walworth).

July 1982

Delivery of Remote Valve Operators (CCI).

September 1983

Delivery of Control Panel

1984

Cycle 6 refueling begins.