Babcock & Wilcox

Power Generation Group

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Telephone: (804) 384-5111

October 11, 1977

Office of Nuclear Reactor Regulation
Attn: Mr. Steven A. Varga
Chief, Light Water Reactors Branch #4
Division of Project Management
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Request for B&W Topical Report on B_4C Absorber Control Rods

Reference: Steven A. Varga to J. H. Taylor, "Request for B&W Topical Report on B_4C Control Rods

Dear Mr. Varga:

This is in reply to the request in the above reference to advise you concerning the B&W schedule for preparation of the subject topical report.

The TVA Bellefonte Units will be the first B&W 205 units that will utilize B_4C control rods. The FSAR for these units has already been prepared and will be filled on December 1, 1977. It is therefore not possible to submit a topical report concerning B_4C one year prior to the submittal of the FSAR on the first plant to use B_4C, as requested, in the above reference.

The TVA FSAR will contain significantly more information concerning B_4C control rods than the B-SAR, since the TVA FSAR conforms to Revision 2 to the SAR format guide. B&W feels that the information provided in the TVA FSAR should be sufficient to meet the needs of a Staff review concerning B_4C control rods. In any event, the use of B_4C control rods will be resolved on the TVA docket and the review should be completed by the end of 1978. This date is well in advance of the schedule for submittal of an FSAR for the first applicant referencing the B-SAR (during the first quarter of 1983).

Mr. Steven A. Varga October 11, 1977

B&W therefore recommends that in lieu of a topical report, the review of B_4C control rods for use in 205 FA units be completed during the licensing of the TVA Bellefonte Units.

Very truly yours,

James H. Taylor
Manager, Licensing

JHT/fw

cc: R. B. Borsum (B&W) Bethesda

RESIDUAL HEAT REMOVAL SYSTEM

1. Valve 751 is, coupled with 750, the RCS (2300 psi) and RHR system interface isolation valves. These valves may not be tested during normal operation since failure of one valve could overpressurize the low pressure RHR system plus their is a protective interlock such that 750 & 751 cannot be opened unless S1-863A&B and S1-862A&B are closed. 862A&B must be open during normal operation or RHR suction is lost.

2. Valve 750 is coupled with 751 to form RCS/RHR interface boundary. Same information applies.

3. FCV-605 is the RHR heat exchanger bypass flow control valve. Section 3.3.1.1 i of the unit Technical Specifications requires this valve to be closed and air supply to the air operator isolated. Valve is in a safety position.

4. HCV-758 is the RHR control valve which controls flow through the heat exchangers. Section 3.3.1.1 i of the unit Technical Specifications requires this valve be closed and air supply to the air operator isolated.

5. 759B is the "B" RHR heat exchanger outlet isolation valve. This valve may be tested during normal operation. Since it is in a high radiation area, remote position verification will be verified during refueling.

6. 759 A is the "A" RHR heat exchanger outlet isolation valve. This valve may be tested same as 759B.

7. 757C, coupled with 757D are the RHR cross connects and can be tested during normal operation.

8. 757D can be tested during normal operation.

METROPOLITAN EDISON COMPANY SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

POST OFFICE BOX 542 READING, PENNSYLVANIA 19603 TELEPHONE 215 – 929-3601

May 11, 1978
GQL 0915

Director of Nuclear Reactor Regulation
Attn: S. A. Varga
Light Water Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station Unit 2 (TMI-2)
Docket No. 50-320
Operating License No. DPR-73

The occurrence, at Crystal River 3, of two separated Burnable Poison Rod Assemblies (BPRA's) has raised the concern that a similar incident might occur at Three Mile Island, Unit 2. Although such an event is not considered likely, based upon the satisfactory performance of other B&W operating reactors, Metropolitan Edison deems it prudent to take certain precautionary measures to provide further assurance that the BPRA's will remain in place. Of the various options available, we have determined that the best course of action is the installation of positive retention devices, which were recommended by B&W. These retainers have been designed and are undergoing test and evaluation at B&W.

Currently, it is our intention to install the retainers on all BPRA's following completion of startup and acceptance testing. As discussed below we are confident that the plant can be operated for up to 75 full flow days prior to installation of the retainers.

TMI-2 NSS operation to date has been with three primary coolant pumps in service. Later portions of the initial startup phase and full power operation will be conducted with all four coolant pumps in service. Crystal River 3 operated for about 300 days with four pump flow before the first indication of BPRA separation occurred. Based on the performance of CR-3 and other B&W 177 Mark B4 Fuel Assembly plants and on wear measurements of the fuel assembly BPRA holddown latches at Crystal River 3, Arkansas Nuclear One-1, Oconee-2, it is conservatively estimated that TMI-2 can reliably operate for up to 75 days of the full four pump flow.

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Results to date of the B&W investigation of the CR-3 event indicate that the separation of BPRA's may be due to a long term wear phenomenon causing separation of the BPRA holddown latch. Coolant flow and the resultant net hydraulic lift compared with the wet weight of a BPRA appears to be a primary factor in the holddown latch wear rate. Latch hardness is also a significant factor. The 68 BPRA's and holddown latches in TMI-2 are of the same design used in all B&W 177 FA reactors.

Analysis of the TMI-2 BPRA hydraulic lift force for four pump flow indicates less nominal lift than at Crystal River 3. The holddown latch assembly minimum hardness on TMI-2 fuel assemblies in also equal to or greater than the hardness of the Crystal River 3 holddown latch assemblies which experienced the highest wear. Thus, TMI-2 can be expected to experience a lower wear rate than Crystal River 3. However, to account for other undefined factors which may influence wear rate, a factor of 4 has been applied to the highest wear rate observed at Crystal River. On this basis, an allowable limit of 75 days of TMI-2 four pump operation has been established.

Wear data from Oconee 2 and ANO-1 for fuel assemblies which operated for as long as 600 full flow days lend confidence that the use of Crystal River wear data coupled with a safety factor of 4 is conservative for TMI-2. Davis Besse with higher calculated lift and comparable minimum holddown assembly hardness has operated without incident for greater than 150 full flow days. Rancho Seco, also with calculated higher lifts but with much higher minimum holddown latch assembly hardness operated for greater than 500 days without incident. Oconee 3 and TMI-1, with calculated lift forces in the same range but slightly lower than TMI-2, both operated for greater than 500 full flow days without incident.

Operation with three pumps precludes BPRA net lift with a very large margin thus avoiding conditions under which wear can occur. To date, all operation in TMI-2 has been with 3 coolant pumps in service. TMI-2 will not be operated past 75 days of accumulated full flow operation, prior to retainer installation, without further justification. The NRC will be informed of the results of any investigations which may change the basis for the allowable period of four pump operation.

Thus, based on the considerations described above, there is a very low probability of a BPRA separating from a fuel assembly in TMI-2 before retainers are installed. These retainers will insure BPRA locking for the remainder of first cycle operation.

In the very unlikely event that a BPRA may become separated during plant operation the consequences to the core are within the bounds of the analysis addressed in the FSAR. Depending upon its location within the core a separated BPRA will have a varying impact upon assembly power peaking. With a significant increase in power peaking the event would be detected by the

tilt alarm or power distribution monitoring and appropriate corrective action would be taken. Lesser power increases would be within the allowable peaking limits considered in the Technical Specifications. In addition, the change in by-pass flow as a result of BPRA removal, is negligible.

The consequences of a BPRA separating from the core are bounded by the results of the Ejected Rod Accident analyzed in the FSAR. The reactivity worth of a single BPRA is only 30 to 40% the worth of a control rod and is less than the maximum ejected rod worth of 0.65% $\Delta K/K$ used in the FSAR. The consequence of a stuck control rod assembly (CRA) is a normal design consideration for calculating shutdown margin. All FSAR accidents are analyzed with the reactivity effects of the most reactive control rod stuck out of the core. The effect of a separated BPRA would be less than a stuck control rod for the same incident.

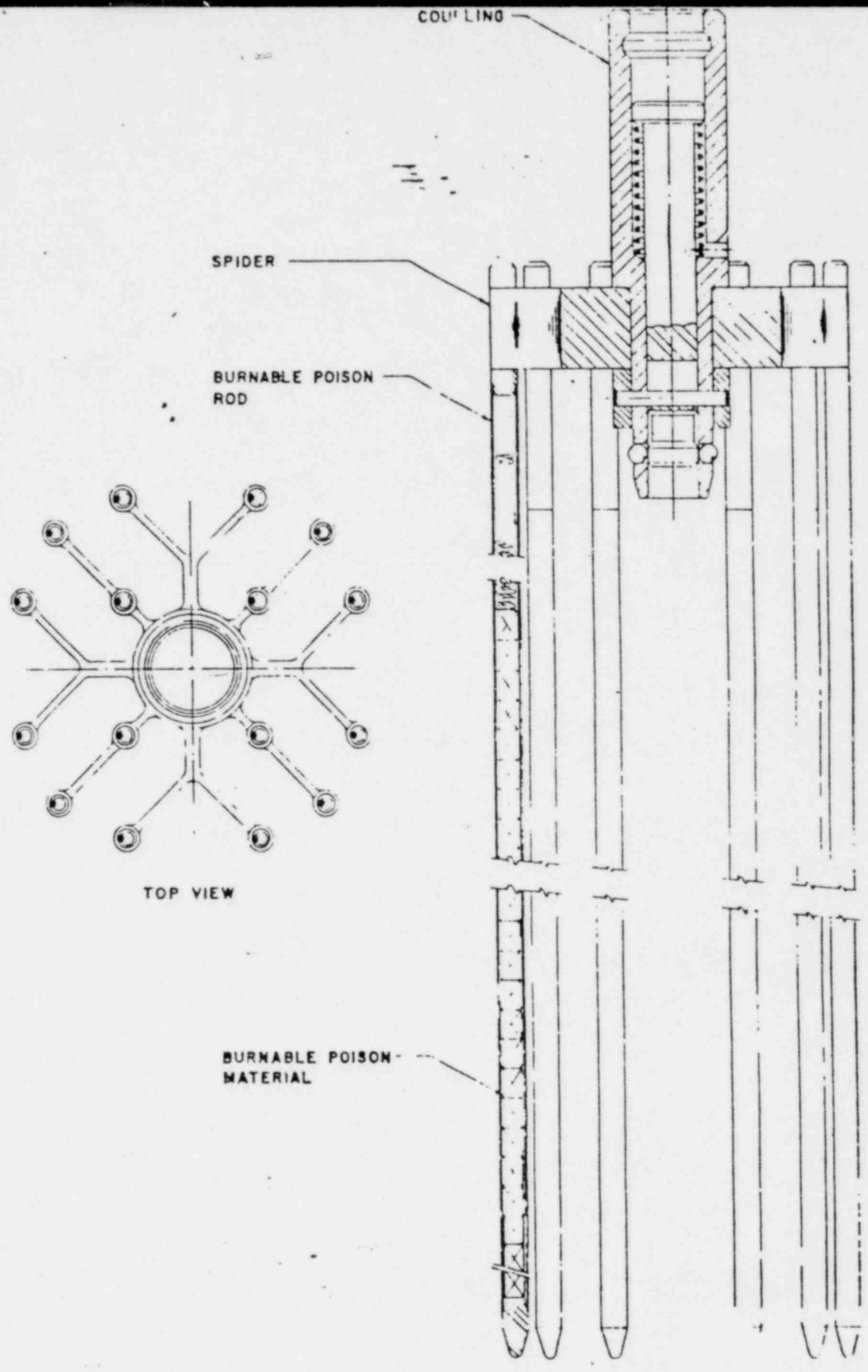
Based upon the above discussion it is concluded that there is a very low probability of a BPRA separating from the core during the limited period of four pump operation; also, any consequences to the core from such a separation are bounded analyses contained in the TMI-2 FSAR.

Sincerely,

J. G. Herbein
Vice President-Generation

JGH:RAL:dkf

cc: H. Silver



BURNABLE POISON ROD ASSEMBLY

THREE MILE ISLAND NUCLEAR STATION U